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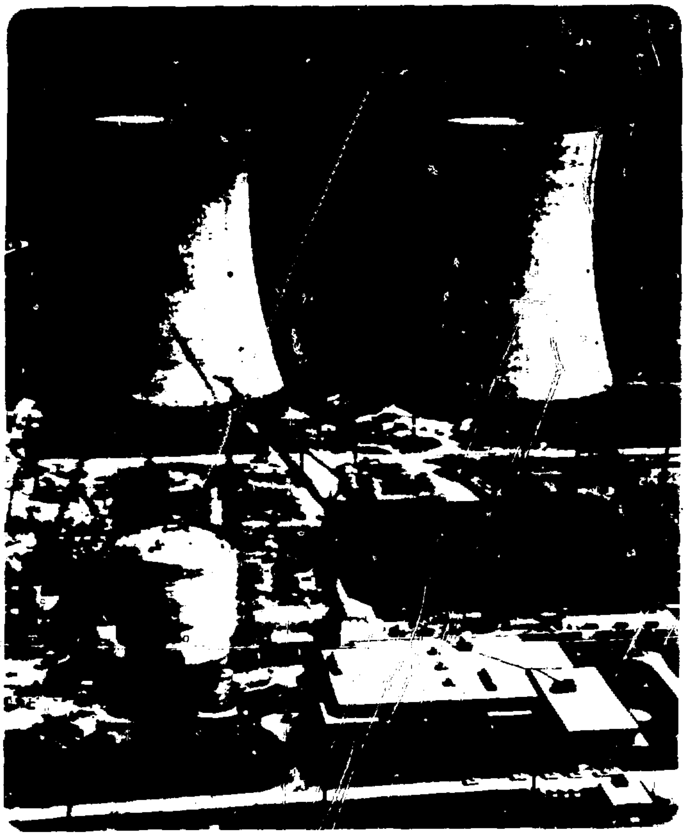

Volume II

Thermal Reactor Safety

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THERMAL REACTOR SAFETY

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April 7-11, 1980

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Date Published: June 1980

FOREWORD

The 1980 ANS/ENS Thermal Reactor Safety Meeting occurred at the time of a major reassessment of reactor safety as a consequence of the Three Mile Island Accident. As stated by Dr. Alvin Weinberg in his brilliant speech closing the meeting, because of modern instant communications, an accident anywhere is an accident everywhere. Thus it is appropriate that safety issues were discussed in the international forum provided by this meeting, thanks to the extensive European participation.

TMI has forced a reevaluation of reactor safety in which the trend is away from the legalistic compliance with rather arbitrary regulatory requirements based on the large loss of coolant accident toward more rigorous assessment of a wider range of more realistic accidents. Fortunately this conference occurred at an ideal time to provide a platform for discussion of many new areas such as risk assessment, human factors, diagnostics, and Class 9 accidents.

We are gratified by the tremendous response from the technical community to our call for papers; although the program was expanded to four parallel sessions we were unable to accept a large number of fine papers.

We are particularly indebted to the members of the technical program committee. They gave us guidance on invited speakers and on areas in which papers should be stimulated, and participated in the grueling paper summary review. We should point out that, consistent with the practice of recent ANS topical safety meetings, 1000-word summaries were reviewed; the authors were requested to provide camera-ready copies of the full papers at the time of the meeting. We feel that this approach, which places trust in the professional responsibility of the author, provides information in a much more timely manner than could be possible under a system of complete review.

We not only wish to thank the authors, but also the projectionists supplied by the University of Tennessee's Nuclear Engineering Department, the organizing and executive committees, the Hyatt Regency Hotel, for their excellent facilities and assistance, and Mrs. Norma Callaham of the ORNL Conference Office for her very capable assistance. Finally we wish to thank the Session Chairmen for their contributions to this meeting.

M. H. Fontana (ORNL)

D. R. Patterson (TVA)
Technical Program Co-Chairmen

PREFACE

The 1980 ANS/ENS Thermal Reactor Safety Meeting was the fourth in a series which includes the 1973 meeting in Salt Lake City, the 1977 meeting in Sun Valley, and the 1978 meeting in Brussels. The 1978 Brussels meeting, as well as the present one, was jointly sponsored by both the American Nuclear Society (ANS) and the European Nuclear Society (ENS). Given the obvious international interest in these last two meetings, I would hope that ANS and ENS will continue to cooperate in sponsoring these meetings.

It is interesting to reflect on these meetings and the predominant theme of each. The Salt Lake City Meeting followed the Emergency Core Cooling Systems' hearings in the U. S. and were dominated by plans for ECCS research. The Sun Valley Meeting was notable for the volume of research results on design basis accidents, the increasing interest in probabilistic risk analysis (WASH-1400) had been issued in 1975, and for the extent of foreign participation. The interest which resulted in the participation was in part responsible for the joint ANS/ENS meeting in Brussels in the fall of 1978. This meeting provided the opportunity for the presentation of European research and development and also reflected the commitment of many European nations to nuclear power development. The present meeting is - not surprisingly - dominated by the repercussions of the March 28, 1979, accident at Three Mile Island-2. This dominance is reflected in research on both sides of the Atlantic Ocean, as well as in an effective defacto licensing moratorium in many countries with the notable exception of France. The 1980 meeting also evidenced increasing interest in risk assessment and, in particular, comparative risk assessment.

In this person's experience, the TMI accident has brought about no philosophical defections among those in the nuclear community who have a comprehensive understanding of risk. There has, however, certainly been an increased perception of the need for handling the more probable nuclear accidents (as opposed to the design basis accident) - as well as a recognized need for improved public education and better public communications. With respect to the public, the media can play an important role and I would hope that they would use their power wisely. We have - in this country and elsewhere - too many short-term solutions for long-term problems.

This meeting owes whatever success it may have achieved, to many persons who unselfishly gave of their time, efforts, and talents. In addition to the members of the program committee, I would like to express my sincere appreciation to the several sponsors and to the various chairpersons who comprised the planning committee. Special thanks are due to Jan B. van Erp of Argonne and Dieter Bünemann of the Institut für Physik (Geestacht) who together combined to make the ENS participation most meaningful. The contribution of the Technology for Energy Corporation of Knoxville, Tennessee, in sponsoring the opening reception is gratefully acknowledged.

Wm. B. Cottrell
General Chairman

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A. R. Buhl - Technology for Energy Corporation

W. Vinck - Commission of European Communities

CONFLICTS ABOUT NUCLEAR POWER SAFETY: A DECISION THEORETIC APPROACH

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ABSTRACT

A series of psychological studies indicate that people's judgments of risks from energy production in general and nuclear power plants in particular deviate from technical and statistical estimates because social and psychological variables influence people's risk perception. After reviewing these studies we will outline a decision analytic methodology which incorporates such social and psychological variables in a formal analysis of the risks and benefits of nuclear energy production. The methodology is intended to identify groups with differing risk-benefit perceptions and to elicit and quantify their values and concerns. Such group and value structures are presented for the problem of choosing between a nuclear plant, a coal plant, and a conservation strategy.

SOME SOURCES OF CONFLICT

The controversy over nuclear energy ranges from technical disagreements about accident probabilities to deeply rooted social conflicts about power concentration and growth. In the center of the debate stands the question: is nuclear power safe enough? Is it an acceptable technology? The following hypothetical discussion between an advocate (A) and an opponent (O) of nuclear power introduces some of the conflicts in that debate.

A: Let me say straight out that I favor nuclear energy, because it is safer, cleaner, and cheaper than any other form of electricity generation.

O: I doubt all three assertion , but especially the statement that nuclear power is safe. Three Mile Island has shown how close we are to a nuclear disaster. In my opinion nuclear power is riskier than any other tehcnology.

A: The statistics prove that you are wrong. Coal mining and burning kills hundreds of people every year. Motor vehicle accidents kill 50,000. In contrast, not a single person has been killed from radiation.

O: Maybe nobody has been killed yet, but what I am worried about is the very real possibility of thousands dying in a nuclear disaster. That threat makes nuclear power riskier than any other technology.

A: You cannot judge a technology merely on the basis of a miniscule probability of an accident. Otherwise we should not be allowed to build dams or fly airplanes. The probability of a serious nuclear accident is so small, you might as well neglect it altogether. In fact, it is smaller than being hit by a meteor. And I doubt that you spend much time thinking about meteors.

O: I am not willing to play any numbers games. I find the attitude of consciously gambling with hundreds or thousands of human lives deplorable, no matter how small the probabilities. What about the dread, the fears, and the mistrust that is generated by such technological gambles? Technical adventures like nuclear power can affect future generations and could shake the very foundation of our society.

A: Don't you think that dependence on Arab oil, unemployment, inflation, brown-outs, acid rain, and CO_2 induced climatic changes -- all possible consequences of stopping nuclear power -- may shake the foundation of our society as well?

O: Why do we depend on Arab oil? Why are we so helpless when it comes to inflation and unemployment? I claim that it is the very philosophy of growth, profit, and power concentration which makes our society so fragile. The alternative to nuclear power is not another 3000 MW electric plant controlled by big business. The alternative is to decentralize energy production and distribution, and to give people more control over the economy.

A: Then you are not merely questioning the safety and economics of nuclear energy production. You are questioning our social system.

O: Yes.

This discussion reveals conflicts about safety of nuclear power on at least four different levels. The initial argument is about facts of nuclear and non-nuclear technology and about different risk estimates. Typically these arguments are about the risk levels expressed either as probabilities of accidents or as degrees of consequences. Such disagreements are not uncommon even within the technical community as the discussion after WASH 1400 has shown [1,2]. Beyond the technical community, factual arguments are often compounded with laymen's mistrust of "facts" or "expert estimates." Still on a relatively technical level, the discussion then turns into an argument about definitions of risk. Most risk analysts express risk in terms of average or expected annual fatalities (or some other consequence measure). Many opponents of nuclear power, on the other hand, concentrate on disasters, ignore probabilities, and stress instead maximum fatalities, dread, and severity of impacts.

On a third level, the controversy is about risk-benefit tradeoffs when comparing energy options. While the advocates stress benefits of nuclear power (resource independence, employment, and growth), the opponents stress the unknown risks (genetic defects) and question the value of the benefits. The fourth and perhaps most fundamental level of argumentation is about the general societal values and ideologies in the evaluation of alternative energy paths. One form of this conflict is the dichotomy between "Small is beautiful" and "Large is necessary;" another dichotomy is "capitalism" vs. "socialism."

While the preceding discussion was illustrative, the following review of the psychological literature will show that the four levels of controversy actually can be observed in empirical studies. After the review we will outline a methodology which aims at defining areas of agreement and clarifying areas of disagreement in the acceptable risk problem. In the final section we will present some initial results of an application of the methodology to the problem of choosing among three electricity supply alternatives in California.

REVIEW OF THE PSYCHOLOGICAL LITERATURE

Slovic, Fischhoff, and Lichtenstein [3,4] conducted a series of experimental studies to determine how technically trained risk assessment experts and politically active laypeople differ when judging risks. In one experiment they asked experts and laypeople to rate the risks of 25 technologies or activities ranging from motor vehicles to nuclear power. Expert risk judgments were essentially identical to best technical estimates of annual average or expected fatality rates. In particular, nuclear power was considered the least risky technology. In contrast, nuclear power was given the highest risk rating by members of the League of Women Voters (LOWV), and by a group of students. In general, laypeople's risk estimates were only loosely associated with technical estimates of annual expected fatality rates.

To determine if laypeople were just misinformed or biassed, Fischhoff et al. [5] asked the same group of League members to give their best estimate of fatalities in an average year. These judgments were quite accurate with a bias to overestimate low rates and underestimate high rates. Most importantly, the same group that judged nuclear power to be the riskiest technology gave it the lowest fatality estimate in an average year.

Why do people's risk judgments differ from expert judgments and from technical estimates of expected fatality rates? Slovic et al. [3] suggest that the factors "disaster potential" and "dread and severity" of a given risk strongly shape laypeople's risk judgments. In fact, the League members' risk judgments were highly (.95) correlated with a linear combination of subjective judgments in the following three factors: expected number of fatalities in an average year, expected number of fatalities in a catastrophic year, and a subjective rating of the dread and severity of the risk. The main differences in the risk judgments of laypeople and technical experts was that laypeople stress disaster potential, while experts stress average fatality rates. As in our illustrative discussion, what appeared to be a disagreement about facts and data (the first level) turned out to be a disagreement about the definition of risk (the second level).

So far, only fatality risks were considered. Otway and Fishbein [6] , and Thomas et al. [7] performed a series of studies which expanded the analysis to a wider spectrum of risk and benefit dimension (the third level of conflict). They utilized Fishbein's attitude measurement model [8] to measure people's beliefs and values with respect to nuclear power. In one study 224 subjects were presented with 39 statements about nuclear power, e.g., "nuclear power is risky" or "nuclear power provides good economic value." Four factors described the structure of the subjects' attitudes towards nuclear power:

1. Psychological risks (involuntary exposure, lack of control, affects large number of people, delayed health effects, changes in man's genetic make-up);
2. Socio-political risks (increased security, weapons proliferation, dependence on experts, terrorism);
3. Environmental risks (water and air pollution, depletion of resources, occupational accidents);
4. Economic and technical benefits (standard of living, economic growth, employment, good economic value, technological innovation).

Subjects in favor of nuclear power strongly believed that nuclear power will lead to economic and technological benefits, while those opposing nuclear power were uncertain about the benefits. Both groups were in agreement that nuclear power carries certain sociopolitical and psychological risks, although the opponents held that belief much more strongly than the advocates.

While the above disagreements were about the subjects' beliefs, there were also some interesting differences about the subjects' underlying values (the fourth level). The advocates of nuclear power favored the industrial way of life and a consumer orientation of society, while the opponents found these aspects negative. Advocates stressed the positive value of national prestige, industrial development, and standard of living, factors which the opponents found less positive. Opponents, on the contrary, favored conservation while advocates were indifferent.

Opinion polls and social surveys indicate that deeply rooted social conflicts may underlie these differences. Anti-nuclear groups and environmentalists often object to a growth philosophy and favor the idea that "small is beautiful." Some anti-nuclear groups also favor political decentralization and socialistic views [.9]. Nuclear advocates and advocates of industrial development, on the other hand, often favor growth, centralization, and the basic principles of capitalism.

We have thus reached what may be the bottom line of the question of risks and acceptability of nuclear power. It appears that social groups strongly differ in their beliefs about risks of nuclear power and in the underlying values driving their judgments of its acceptability.

METHODOLOGICAL APPROACH

The above review indicates that methodologies for aiding decision processes on nuclear safety are likely to fail if conflicting perceptions and opinions are not addressed, if psychological variables are ignored, and if social and political value differences are denied. The present methodology development therefore attempts to include psychological and social variables in an analysis of the risk and benefits of nuclear power. Decision analysis and multiattribute utility measurement [10,11] are used as a starting point for the methodology development, since they can provide the necessary comprehensiveness, and since they can cope with highly soft and judgmental variables.

As a decision making context for the methodology development we choose the "Notice of Intention" (NOI) process of the California Energy Commission. All California electric utility companies are required to file an NOI when planning to build a new power plant. As part of the NOI process the Energy Commission has to determine whether alternatives to the planned addition exist which are preferable on the grounds of environmental, safety, economic, and other considerations. Such determination obviously requires a consideration -- formal or informal -- of a spectrum of safety issues, and of risks and benefits of the proposal and its alternatives. In our methodology development, we consider three main possibilities: nuclear, coal, and a conservation strategy with addition of "soft" capacities. The ruling of the Commission is made based on a series of investigations and hearings. In these hearings a number of political, environmental, and industry groups voice their opinions about the economic, legal and environmental aspects of supply alternatives.

A multiattribute evaluation is performed for each group separately to quantify the risk and benefits of the supply alternatives. On this level there is no need for inter-group consensus. Inputs into the analysis are each group's evaluation criteria and each group's best estimates of the performance of the options on these criteria. Comprehensiveness is stressed rather than detailed assessment of probabilities and consequences. All this is done in the usual decision analysis methodology which allows substantial shortcuts in estimating impacts and probabilities by judgmental methods.

Outputs of the analysis are group specific risk-benefit indices for the supply alternatives. The analysis can then be used for a number of purposes: to identify areas of agreement and disagreement on any of the four levels of conflict described earlier; to determine additional information needs where conflict is about data and expertise; to discuss disagreements about measurement definitions (e.g., "fatality risks") and attempt to resolve them; to determine if incremental changes in supply alternatives (e.g., remote siting) can produce more acceptable solutions. Of course, no analysis can take away the ultimate responsibility of the lead agency to make its ruling.

Developing the methodology outlined above is a very ambitious task. The effort is planned for a two year period, including some testing and application of parts of the methodology. The development is done in five steps:

1. Identification and structuring of groups concerned with risks and benefits from the nuclear-coal-conservation decision;
2. Adaptation of decision analytic value tree techniques to structure the relevant values and concerns of these groups;
3. Experimental research to construct scales and criteria that can measure group specific values;
4. Adaptation of multiattribute utility measurement techniques for technical and judgmental measurement of the energy options on these scales;
5. Adaptation of decision analytic evaluation and inference models to aggregate such measurements to groups specific risk and benefit estimates.

SOME RESULTS

The structuring steps (1 and 2) of this methodology development have been completed. The first results are generic structures of groups involved in the coal vs. nuclear vs. conservation decision, and specific structures of groups involved in the California NOI process. Table I presents some results for the NOI process. In our interveiws of the groups listed we were usually able to identify the levers and decisions for which that group is responsible in the NOI process. We also could tentatively identify the group's rank ordering of the three options (coal, nuclear, conservation). However, it is much more difficult to determine the underlying values that drive the rank ordering. In the last column of Table I we list our impression of the main values that in our opinion drove the preference ordering of the respective group.

The next step was to examine the values and concerns of some of the groups in some more detail. We therefore began to build value trees for selected groups. Value trees put together in a hierarchical organization, general goals and objectives, intermediate objectives, and specific criteria for the comparison of risks and benefits of the three options. Value tree structuring is a recursive and ongoing process. We based our initial value trees on interviews and discussions with representatives of the groups with which a tree is built. We are now in our second stage of iteration in building value trees for a utility company, for an environmental group, and for an anti-nuclear group. Table IIa shows the superstructure of the value tree that might represent a utility company's point of view when comparing the coal-nuclear-conservation options; Table IIb might represent an environmentalist's group's point of view. The full trees are, of course, much more complex, extending down to about 5 layers and ending up with up to 50 twigs. Even in the superstructure differences are obvious. While the environmentalists' tree directly addresses environmental values, the utility's tree considers environmental concerns mainly through the legal requirements for licensing. We are also developing a tree for a politically active anti-nuclear group. This tree seems to have yet another super-structure, stressing health, quality of life and sociopolitical considerations. "Pure" environmental (soil erosion) and cost (investment) considerations seem to play a lesser role.

We can also detect some differences in the actual definitions of variables such as "cost" and "health risk." Environmentalists and anti-nuclear groups tend to operationalize "cost" of energy production in terms of <u>all</u> direct and indirect costs, including, for example, governmental outlays in R&D and waste disposal. Cost calculations by utility companies usually include only the more tangible direct costs. Our interviews also confirm the differences in the definitions of "risk." Environmentalists and anti-nuclear people indicated that risk is a much more multifaceted concept than expected fatalities.

In the next months, we will continue to refine the trees for the utility company, for an environmental group and for an anti-nuclear group through continued structuring interviews with several group representatives. We will then enter into the actual quantification steps 3-5.

ACKNOWLEDGEMENT

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TABLE I

LEVERS, PREFERENCES AND VALUES OF SOME GROUPS INVOLVED IN THE NOTICE OF INTENTION PROCESS

GROUP	LEVERS IN NOI	PREFERENCES	BASIC VALUES/CONCERNS
Utility Company	Propose facility and site Defend proposal	1. Nuclear 2. Coal 3. Conservation	Service Financial stability Acceptability
Energy Commission	Approve/disapprove need, facility, site	1. Cons./Cogeneration 2. Coal 3. Nuclear	Conservation Flexibility Environment and safety
Bankers and Investors	Underwriting, financing, lending	1. Coal 2. Nuclear 3. Conservation	Financial soundness of project and company
Environmentalist	Intervenor, advisor	1. Conservation 2. Coal 3. Nuclear	Protect flora and fauna Environmental protection
Anti-Nuclear groups	Direct action, demonstration	1. Conservation 2. Coal 3. Nuclear	Long term impacts on society, political concerns

Table IIa

POSSIBLE VALUE TREE SUPERSTRUCTURE OF THE "UTILITY COMPANY"

- Project Cost
    - Investment
    - Operation/maintenance
    - Fuel

- Service Improvement
    - Reliability
    - System stability
    - Supply mix

- Company Finances
    - Stock value
    - Sales
    - Return on investment

- Project Feasibility
    - Licenseability
    - Political acceptability
    - Financial feasibility

- Environment and Safety at the Margin
    - Environment
    - Safety

Table IIb

POSSIBLE VALUE TREE SUPERSTRUCTURE OF THE "ENVIRONMENTALIST"

- Health and Safety
    - Mental health
    - Physical health

- Environmental Impacts
    - Flora and fauna
    - Water and air
    - Land and soil
    - Climate
    - Depletion of resources

- Impacts on Lifestyle
    - Aesthetics
    - Recreation
    - Culture
    - Conveniences

- Socio-Political Impacts
    - Growth
    - Resource independence
    - Flexibility, resilience
    - Potential for terrorism
    - Potential for proliferation
    - Equity of risk and benefits

- Monetary Energy Cost
    - To the utility company
    - To the taxpayer
    - To the resident
    - To the ratepayer

OYSTER CREEK PROBABILISTIC SAFETY ANALYSIS (OPSA)

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ABSTRACT

Highlights are presented of an independent probabilistic risk assessment of the Oyster Creek Nuclear Power Plant. The study was conducted between November 1977 and August 1979 for the Jersey Central Power & Light Company. Elements of the study included release frequencies, common cause analysis, and consequence analysis. While the study was based on WASH-1400 methodology, advances in risk analysis were used in quantifying uncertainty and modeling site specific characteristics. The study provides a basis for evaluating the impact on risk of plant modifications and procedural changes.

INTRODUCTION

A comprehensive risk analysis was performed on the Oyster Creek Nuclear Power Plant.(1) Oyster Creek, a boiling water reactor, has been in commercial operation since December 1969. The purpose of OPSA was to:

- Provide an independent and quantitative analysis of the level of risk associated with the continued operation of Oyster Creek.
- Provide a framework within which to evaluate the impact on risk of proposed plant changes.
- Train utility staff in the application of formal risk analysis.

It is the purpose of this paper to highlight the methods employed in OPSA and to share some preliminary results. Final results are not available as OPSA is still in the draft stage. The elements of OPSA and selected outputs and topics are illustrated in Figure 1. Figure 2 is a breakdown of events contributing to core damage and served as a guide for organizing the release segment of the study. The overall output of OPSA resulting from combining the release frequencies and consequence analyses was the probability of frequency of different levels of damage to the public.

OPSA adopts an extension of the methodology of the Reactor Safety Study.(2) The differences center around four principal areas. They are: (1) the details of event tree construction, (2) methods for handling data, (3) seismic analysis, and (4) the model for the consequence analysis.

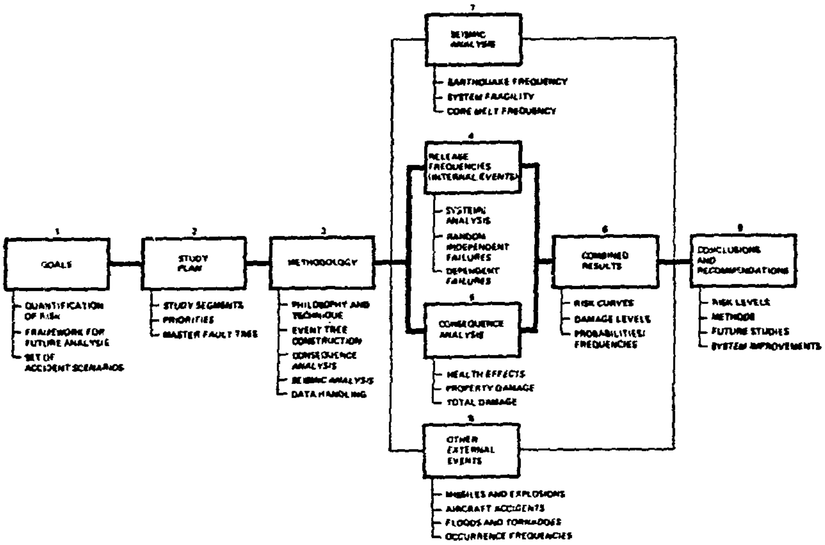


FIGURE 1. ELEMENTS OF OPSA

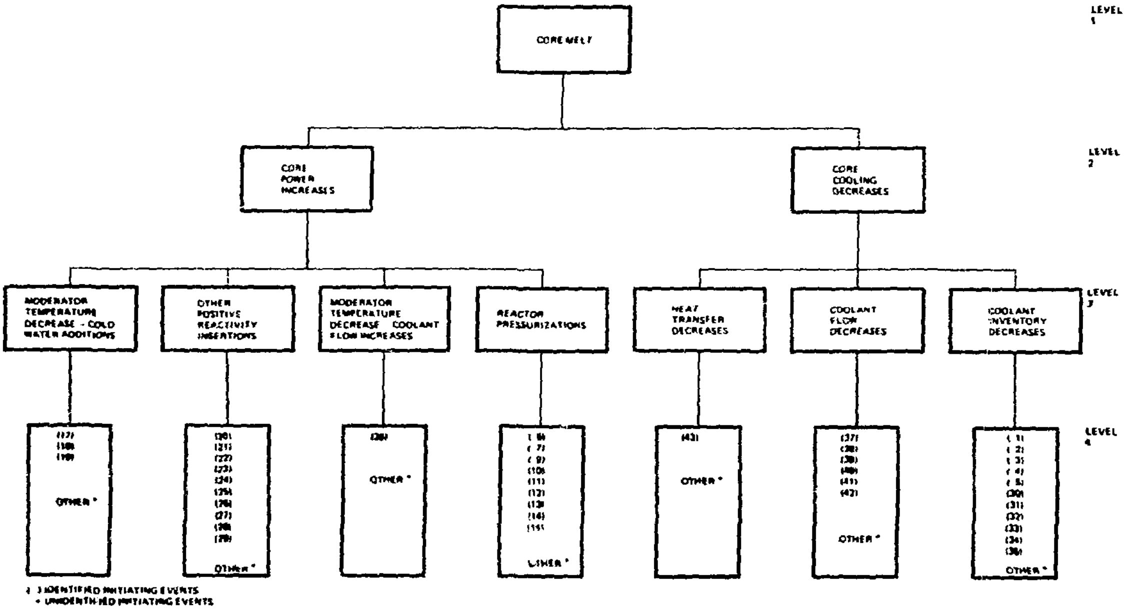


FIGURE 2. OYSTER CREEK NUCLEAR STATION RISK LOGIC MODEL

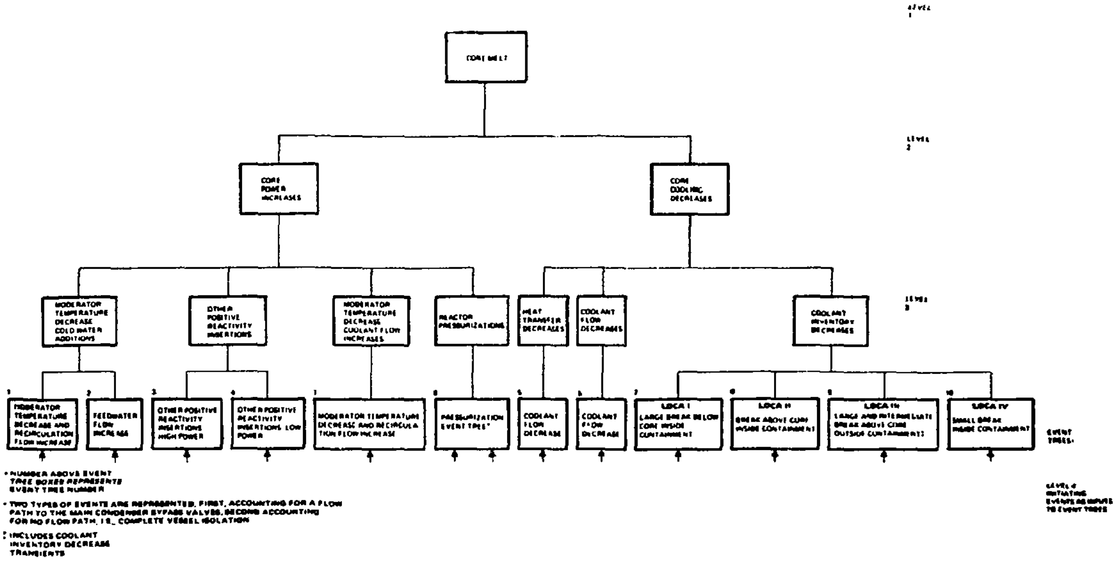
The primary reasons for the differences are related to:

- The plant specific, site specific qualities of OPSA.
- Advances in risk analysis since the Reactor Safety Study.

RISK ANALYSIS AND EVENT TREES

A risk analysis in its most basic form is a set of accident scenarios, their probability of occurrence, and their impact.' The scenario format adopted was the event tree. A distinguishing feature of OPSA was the level of detail of the event trees. Detailed event trees were judged as the best approach for a more visible treatment of plant behavior. The event tree approach enhanced the involvement of plant design, operations, and maintenance personnel. The result was greater assurance that the important accident sequences were considered. In general, fault trees were used to investigate the frequency of failure to start and failure to run of mitigating systems constituting the branch points of the event trees.

Judicious construction of the event trees lead to ten trees representing over five million scenarios. The approximately five million scenarios were reduced to just under 1,500 for detailed investigation. Finally, the risk was dominated by only a very few scenarios. The relationship of the ten event trees to the overall study is illustrated in Figure 3.



• NUMBER ABOVE EVENT TREE BOXES REPRESENTS EVENT TREE NUMBER

* TWO TYPES OF EVENTS ARE REPRESENTED. FIRST, ACCOUNTING FOR A FLOW PATH TO THE MAIN CONDENSER BYPASS VALVES. SECOND ACCOUNTING FOR NO FLOW PATH, I.E., COMPLETE VESSEL ISOLATION

‡ INCLUDES COOLANT INVENTORY DECREASE TRANSIENTS

FIGURE 3. RELATIONSHIP BETWEEN OYSTER CREEK RISK LOGIC MODEL AND EVENT TREES

DATA HANDLING

The "state of knowledge" probability approach was used throughout OPSA to express knowledge of failure rates, repair times, and safety related events. In general, the state of knowledge of the variable x was expressed as a discrete probability distribution. A probability arithmetic was established to permit such operations as addition, subtraction, and multiplication of discrete probability distributions. Thus, the full state of knowledge was propagated through the event/fault trees to establish the risk curves. In many cases the method of moments was used to propagate uncertainty. That is, the mean and variance of selected distributions were found analytically and propagated through the accident scenarios to determine frequencies of release. Finally, the specialization of data to the Oyster Creek plant was achieved by the use of Bayes' theorem.(3)

The two main sources for generic data were the Reactor Safety Study and IEEE STD-500.(4) In special cases (e.g., rupture of the torus) it was necessary to poll experts and construct a histogram for the failure rate.

SEISMIC ANALYSIS

The seismic analysis methodology used in OPSA can be viewed as consisting of four main stages. The first is the stage of seismology and involves expressing the knowledge of the frequency with which seismic events of various sizes occur at the Oyster Creek location. The second, or structural stage, relates to the response of the plant structures and internal components to earthquakes of various magnitudes. In particular, this is the stage to determine the likelihood of failure under the defined earthquakes of various items of safety related equipment and structures.

Given the failure likelihoods of individual equipment and structures, the third stage consists of determining the likelihood of core melt and release under various quakes. The fourth and final stage is the assembly of the information into an earthquake risk curve in probability of frequency format.

The event trees were used to determine the frequency of core melt due to earthquakes. Important features of the analysis included the treatment of uncertainty and dependent failures. No results of the seismic analysis are presented at this time as considerable rework is anticipated before the final report.

CONSEQUENCE ANALYSIS

The consequence analysis of OPSA was based on a modification of the Reactor Safety Study Consequences Computer Program (CRAC).(2) As used in WASH-1400, CRAC models both the plume travel and evacuation path in straight lines extending radially outward from the plant. To utilize the program at Oyster Creek, modifications to CRAC were made to answer questions about the importance of site specific features. Examples of features of concern include sea breeze (and other phenomena which may cause persistent plume direction changes), and geographical considerations which could strongly influence evacuation paths. In particular, the changes to CRAC were the use of variable direction plume trajectories and a variable direction (and speed) evacuation scheme. The modified version of CRAC resulting from OPSA is referred to as CRACIT (for calculation of reactor accident consequences including trajectories).

Six different types of consequences were selected for specific evaluation in OPSA. They were early fatalities, early injuries, thyroid cancers, other cancers, population total body dose and property damage. Modeling the peculiarities of the Oyster Creek site was enhanced by the ability to input meteorology data from three different locations. Complementing the excellent source of data concerning the site were two other sources; McGuire Air Force Base and LaGuardia. Assumed accident start dates were selected randomly in sets stratified so that start times are uniformly distributed over all months, and each month has the same number of day and night samples. The radiation dose calculation portion of the CRACIT consequence model has not been significantly modified from that used in WASH-1400.

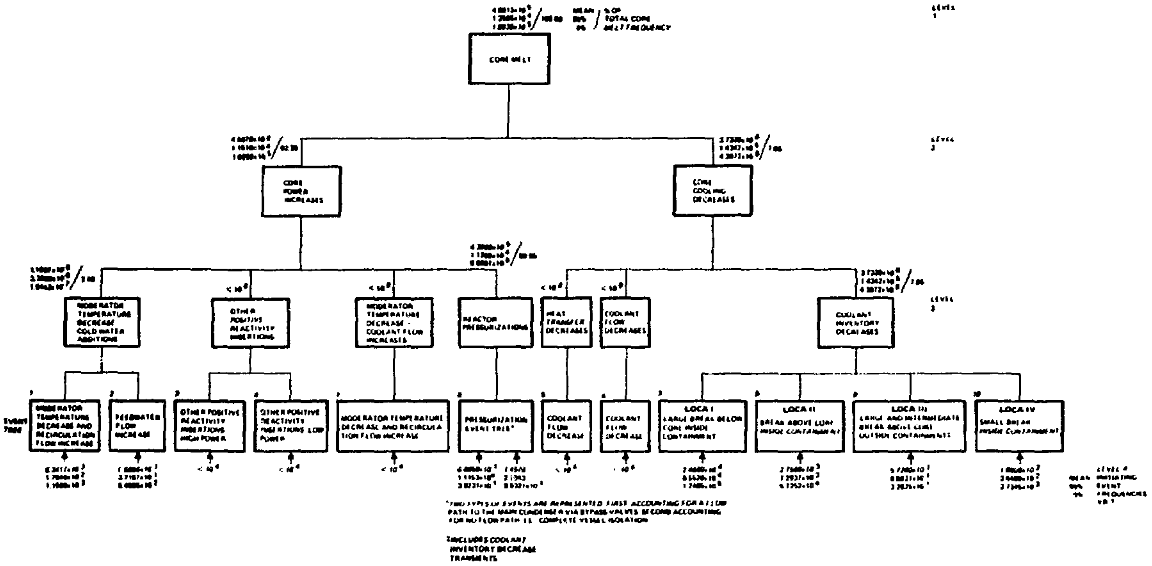
CRACIT calculates sets of consequences from all of the combinations of release categories and random samples of meteorological conditions. Each consequence set has a frequency of occurrence associated with it which is defined by the frequency of occurrence of a release falling into a given release category, multiplied by the probability of the effect based on a set of meteorological conditions. After the frequencies of all consequence sets have been calculated, they are combined to yield an overall frequency distribution of consequences versus probability. The results (effects) from each meteorological sequence for each release category are combined to form cumulative probability distributions similar in form to those in WASH-1400.

RESULTS AND CONCLUSIONS

It is anticipated that several key analyses contained in OPSA will be expanded prior to completion of the final report. These include seismic and consequences. Thus it is premature to discuss their results. It is possible to discuss the release frequency analyses as a result of internal events - the type of events noted in Figure 3.

The frequency distributions of the initiating events and mitigating systems are propagated through the event trees to obtain the release frequencies associated with the accident sequences. The frequency distributions are represented by means and variances.

Individual accident sequence frequency distributions are appropriately combined to obtain core melt frequency distributions at the event tree level. Total core melt frequency is obtained by combining the individual event trees. The order of the process and some results are given in Figure 4.



*TWO TYPES OF EVENTS ARE REPRESENTED: FIRST ACCOUNTING FOR A FLOW PATH TO THE MAIN CONDENSER VIA BYPASS VALVES, SECOND ACCOUNTING FOR NO FLOW PATH I.E. COMPLETE VESSEL ISOLATION

†INCLUDES COOLANT INVENTORY DECREASE TRANSIENTS

FIGURE 4. QUANTIFIED OYSTER CREEK RISK LOGIC MODEL

Adopting the BWR release categories of WASH-1400 leads to the results in Figure 5. Figure 5 can be viewed as a risk curve in histogram form where damage is interpreted in terms of release category. The means, 5% and 95% bounds for OPSA and WASH-1400 are shown. The analyses leading to these results enable the identification of the dominant sequences in terms of contribution to risk. While the results are preliminary, it is obvious that a small number of sequences dominate the risk. In particular, just eleven sequences out of the 1500 examined account for over 90% of the total core melt. Among the dominant sequences are:

- Reactor pressurization following a sudden inadvertent reactor vessel isolation and failure of the reactor scram system .

- Reactor pressurization caused by a full reactor vessel isolation with subsequent successful reactor scram and a coincident loss of all DC power.

- Reactor pressurization caused by a partial reactor vessel isolation, a failure to scram, and a failure of relief valves to reset resulting in containment overpressure.

- A small LOCA inside the containment with a subsequent loss of coolable fuel geometry.

- A large LOCA with a subsequent loss of coolable fuel geometry.

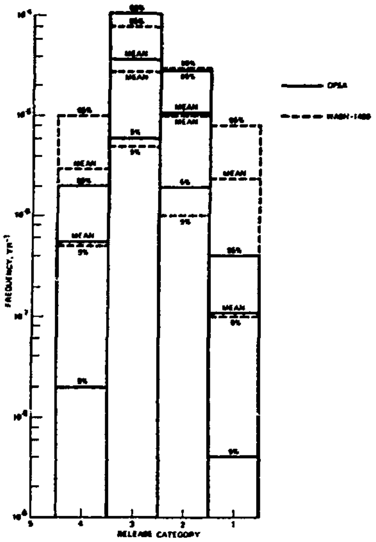


FIGURE 5. CORE MELT FREQUENCY DISTRIBUTIONS
FOR RELEASE CATEGORIES IN HISTOGRAM FORMAT

The combined contribution of the two LOCA sequences contributes less than 5% of the total core melt frequency.

The dominant system contributing to risk is the scram system. In fact, the scram system is many times more important than any other system or function. Other contributors in order of decreasing importance are coolable geometry, DC power supply, relief valves closing, fire pond connection to emergency core cooling, isolation of break outside of containment and feedwater system. It should be noted again that these results do not include contributions from events such as earthquakes, fires, and sabotage.

Among the reasons that the scram system ranks high as a relative contributor to risk is that there is considerable uncertainty in its failure rate. It is worthwhile to briefly outline the analysis performed on this key system.

The important steps in the analysis were the fault tree construction, the analysis of dependent failures, the experiential update, and the construction of the final failure probability curve. To investigate common cause or dependent failures, five subsystems were analyzed. The systems were sensors, logic, hydraulic control units, control rod drives, and scram discharge volume. A discrete distribution for total dependent failure frequency was developed for each of the five systems. The histograms of the five subsystems were combined to yield the discrete form of the prior distribution of the frequency of failure to scram due to dependent failures.

Comparing the results of the independent failure analysis with the results of the dependent failure analysis clearly indicated the dominance of the latter. Finally, consideration of all the available experience data and updating the state of knowledge accordingly led to the final results given in Figure 6 and Table I. The final composite curve expresses the state of knowledge of the OPSA team at this time. The distribution can be summarized by the following values: 5th percentile: 6×10^{-6}; median: 2.8×10^{-5}; mean: 5.4×10^{-5}; 95th percentile: 1.2×10^{-4} per demand.

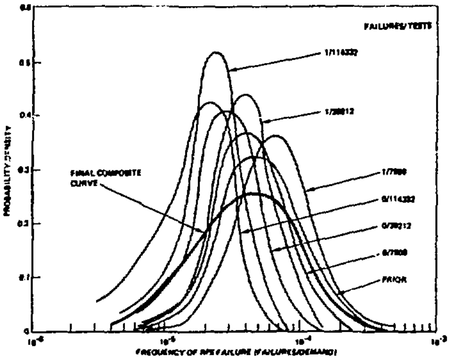


FIGURE 6. FINAL COMPOSITE PROBABILITY CURVE FOR SCRAM FAILURE RATE

TABLE I. INTERVAL PROBABILITIES FOR FINAL COMPOSITE CURVE

Frequency Interval	Probability
3.16×10^{-6} – 5.62×10^{-6}	0.013
5.62×10^{-6} – 1.00×10^{-5}	0.064
1.00×10^{-5} – 1.78×10^{-5}	0.124
1.78×10^{-5} – 3.16×10^{-5}	0.208
3.16×10^{-5} – 5.62×10^{-5}	0.250
5.62×10^{-5} – 1.00×10^{-4}	0.214
1.00×10^{-4} – 1.78×10^{-4}	0.098
$.78 \times 10^{-4}$ – 3.16×10^{-4}	0.027
3.16×10^{-4} – 5.62×10^{-4}	0.002
	1.000

While not all to the same detailed level, a similar analysis was conducted on 16 other mitigating systems important to risk.

The ability to rank initiating events, sequences and systems in terms of contribution to risk supported detailed recommendations on procedural and system changes. Conversely, the framework leading to specific recommendations provided a basis for evaluating the impact on risk of proposed plant modifications or changes in procedures.

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SYSTEMS INTERACTION METHODOLOGY DEVELOPMENT

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ABSTRACT

A program to identify and evaluate potential systems interactions in pressurized water reactors was conducted for the NRC. Fault trees were developed for three functions needed to prevent core damage: loss of the reactor coolant pressure boundary, failure to remove decay heat, and failure to achieve or maintain reactor subcriticality. The objectives were to develop a methodology with criteria for revealing important systems interactions and to assess the Standard Review Plan (SRP) for completeness regarding systems interactions. Computer analysis of the fault trees using linking characteristics between system components resulted in minimal cut sets which were then studied for important interactions. It was found that the SRP did not specifically address: actuation and location of pressurizer relief and isolation valves, cooling and location of auxiliary feedwater pumps, and power to the pressurizer heaters. Application of the methodology to an exemplary plant indicated that the types of systems interactions within the program scope were not a problem.

INTRODUCTION

The Systems Interaction Methodology Applications Program was intended to be a contributing element to the resolution of the problem being addressed by the NRC Task Action Plan A-17[1]. The objectives of the program were to develop a methodology independent of the Standard Review Plan (SRP)[2] for identifying and evaluating systems interactions which affect the likelihood of core damage in light water reactor commercial power plants, and to assess the SRP to determine its completeness regarding systems interactions.

The scope of the study was restricted to allow the methodology to be developed and demonstrated in a timely fashion. The basis for the program was a single unit of a Westinghouse Pressurized Water Reactor[3] under normal environmental conditions. The undesired top event considered was unacceptable reactor core damage. The spent fuel pool and radwaste system were not studied.

There are three plant functions which clearly contribute to the top event: loss of the reactor coolant pressure boundary, failure to remove decay heat, and failure to achieve or maintain reactor subcriticality. Fault trees for each of these three functions were developed for all plant modes except refueling, and for normal shutdown operations and incidents of moderate frequency[4]. Each fault tree model applies to the plant being in any one of five operating modes at the time it was challenged by any one of seven occurrence categories as follows:

Power Operation
Startup
Hot Standby
Hot Shutdown
Cold Shutdown

for

Loss of Offsite Power
Loss of the Power Conversion System Condenser
Normal Shutdown
Inadvertent Moderator Cooldown
Inadvertent Rod Withdrawal
Inadvertent Reactor Coolant System Dilution
All other ANS-51.8/N18.2 Condition II Occurrences

A systems interaction of interest is an event or sequence of events causing two or more components to fail to perform their function, thus increasing the likelihood of an undesired event. Components may act upon one another or be subjected to a common failure. They may be in different trains of the same system or in different systems contributing to the same function. A system interaction may be characterized by the cause initiating the interaction and the connection which permits the interaction.

Possible causes include hardware failures, human errors, and external energies. Human errors may be a cause as well as a connection. Possible connections are categorized as physical, spatial, inherent, and human. Physical connections, as the name implies, involve a material connection between components such as electrical, mechanical, or hydraulic. Spatial connections have various causes which are transmitted through space and thus result from common locations defined by those causes. Inherent connections relate to such factors as the same manufacturer or similar components in redundant trains. These causes and connections may be combined to establish linking characteristics. Six such linking characteristics were considered in this analysis: motive power, control power, actuation, cooling, lubrication, and location. The methodology is believed to be applicable to other areas not within the scope.

FAULT TREES

The purpose of the fault trees was to model the combinations of components which, if failed, would result in loss of any of the three functions and by assumption result in possible unacceptable core damage. These fault trees become vehicles for the identification and evaluation of system interactions which could lead to degradation of plant safety. Support systems, such as ac power, dc power, service water, and compressed air are included as linking characteristics in the analysis. The methodology developed in this program purposely avoided developing support systems not contributing to important systems interactions within the scope.

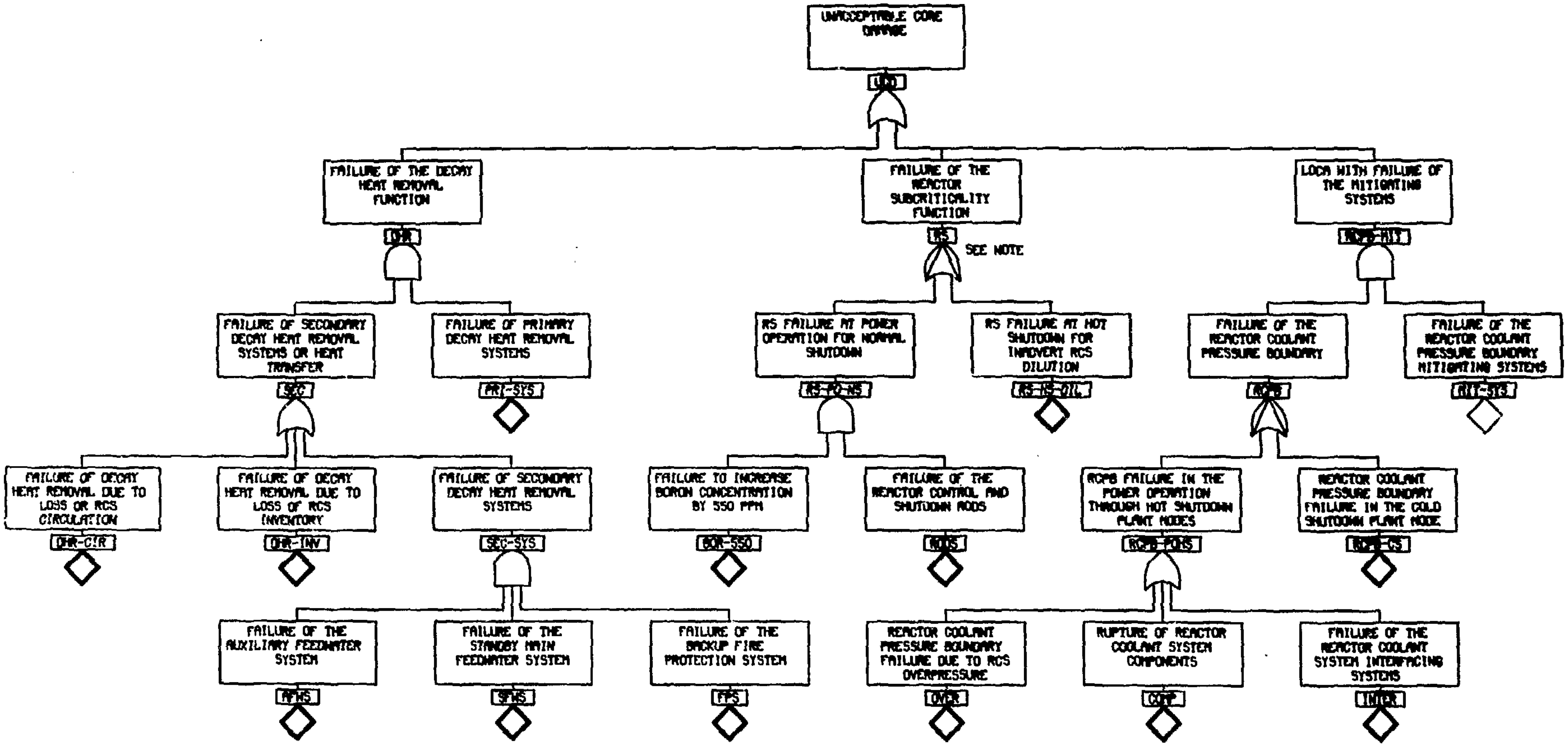
A simplified fault tree starting with the top event is shown in Figure 1. Failure of the decay heat removal function results given failure of all applicable primary and secondary systems. The applicability of the systems is determined from conditions imposed by the initiating occurrence and the plant mode at that time. There are six possible combinations of these systems that can be constructed from the actual fault trees, depending on the plant mode and occurrence category. This is not obvious from the simplified tree, but is depicted by notes on the actual tree. In the worst case, loss of offsite power when the plant is in some mode above cold shutdown, only the Auxiliary Feedwater System is available and thus the analysis was concentrated there. This analysis included the loss of Reactor Coolant System (RCS) circulation and loss of RCS inventory, both of which would prevent the transfer of heat from the primary to the secondary.

Failure of the reactor subcriticality function was divided into ten plant mode/occurrence category combinations; only two are shown in Figure 1 with one developed to an additional level. Essentially, each case involves the potential use of boration or control and shutdown rods to insert negative reactivity. The specific negative reactivity required was derived from the initial conditions. The applicable boration components and flow paths depend on the required change in boron concentration.

Failure of the reactor coolant pressure boundary was modelled separately for cold shutdown and the other plant modes. Although the mitigating systems must also fail in order to result in core damage, this was not part of the scope requested for Phase I of the program. Nevertheless, potential systems interactions may be observed in developing the loss of the reactor coolant pressure boundary, especially due to overpressure. Overpressure may result after the initiating occurrence given an additional equipment failure not related to the initiating occurrence and failure to control the pressure.

ANALYSIS

The fault trees were analyzed to determine the susceptability of the plant to potential systems interactions which could prevent or degrate the performance of the plant function. The Set Equation Transformation System (SETS)[5], a computer code which performs Boolean algebra manipulation, was used to derive the minimal cut sets for the fault trees. A minimal cut set is a Boolean reduced product



NOTE: Only two of the ten combinations of operating mode and occurrence category are shown.

FIGURE 1. Simplified Fault Tree for Unacceptable Core Damage

of events that leads to the undesired top event. The SETS code determines all the minimal cut sets (i.e., no truncation was used based on probability values or number of events in the cut set). This results in a large number of cut sets which were subsequently (after the linking characteristics were assigned) reduced to a smaller number of cut sets having the most significant systems interactions. The main criteria for this reduction was that a potential interaction had to occur in a cut set consisting of less than four independent events.

The methodology is based on the premise that potential systems interactions can be found by identifying commonalities between the components of the systems. These commonalities are described by the linking characteristics previously defined.

The generic analysis was used to assess the SRP including its supporting documents. Table I illustrates the linking characteristics used for the generic analysis as well as their application to specific components. For the generic analysis, all the components were given a complete set of linking characteristics although it was realized that for a particular plant specific components of that type may not exhibit the complete set. The computer was then used to implement a transformation

TABLE I. Linking Characteristics for Various Components

	Linking Characteristics								
	Motive and Control Power								
Component	ac Power	dc Power	Air	Steam	Hydraulic	Actuation	Cooling	Lubrication	Location
Motor Driven Pump	X	X				X	X	X	X
Turbine Driven Pump	X	X		X		X	X	X	X
Motor Operated Valve	X	X				X			X
Air Operated Valve		X	X			X			X
Hydraulic Valve	X	X			X	X			X
Manual Valve									X
Safety Valve									X
Relief Valve	X	X	X			X			X
Check Valve									X

of variables which relates the component events to their respective linking characteristics. The cut set equation was then manipulated to produce a new set of cut sets in which the independent events are component events and/or linking characteristics[6] The new cut sets were then ordered with those having the least number of independent events (component events or linking characteristics) ranked highest. The most significant potential interactions are those which involve all the events of a cut set. This would appear as a single element cut set in which the single element was a linking characteristic.

The SRP is for the most part written in general terms (e.g., systems rather than components). For this reason all the information attained through the sorting techniques for the generic case was coalesced into broader categories. Questions were formulated which could be used in the SRP review. An example of such a question is: "Does the SRP and its supporting documents assure that the power operated relief valve and its associated isolation valves will not share a common actuation signal?" The basic approach for the SRP review was to first review the basic system SRP sections for requirements which would cover the concerns expressed in the questions. Other documents including other SRP sections, Branch Technical Positions, General Design Criteria, Regulatory Guides, IEEE Standards and sections of the ASME code were also reviewed. The results may be categorized as follows:

(1) No statement was found that addresses the question,
(2) general statements were found that imply coverage (e.g., single failure criterion), or
(3) a specific statement was found which is intended to provide the desired coverage.

Finally, it was necessary to make a judgement of importance of the potential interaction. This included consideration of the number of events (component failures or linking characteristics) in the cut set, the likelihood that the linking characteristic could actually cause all linked components to fail, the likelihood of the system failure relative to the likelihood of the potential interaction, and the relative importance of the plant function to public safety.

The specific analysis was based on a specific exemplary facility and deals with much finer detail. Each component was analyzed to determine its actual supporting systems using the same categories of linking characteristics in the generic analysis. The same plant functions were analyzed.

RESULTS

The principal result of this study is the development of a systematic and disciplined methodology for facilitating the identification and evaluation of a range of potential systems interactions. The method has been demonstrated on a meaningful subset of those interactions. The methodology's greatest utility is in providing a systematic method for focusing review on important areas where the potential for interactions may exist which could affect the likelihood of core damage. Therefore, the methodology facilitates thoroughness in the review and conserves resources.

The methodology was applied to an exemplary facility to achieve two goals: (1) to provide a basis for the SRP review and (2) to demonstrate the methodology.

In general, it was concluded that application of the methodology should not be limited to those systems explicitly identified in the SRP as "safely related." In addition to this general conclusion, several "soft spots" were identified in the SRP. These met all of the following criteria: (1) a potential cause of an interaction could be identified, (2) if it occurred, it would increase the likelihood of core damage, and (3) the potential was not explicitly covered in the SRP.

These soft spots were the absence of explicit assurances in the SRP or its supporting documents that: (1) the reactor coolant pressure boundary integrity will not be lost as a result of interactions stemming from a common location or common actuation of the pressurizer power operated relief valves and their isolation valves, (2) the decay heat removal function will not be lost as a result of interactions stemming from a common location or common cooling between trains of the auxiliary feedwater system, (3) positive pressure control will not be lost as a result of interactions stemming from common power sources between pressurizer heater channels, and (4) the inventory makeup necessary to maintain decay heat removal will not be lost as a result of interactions stemming from the common location of the Refueling Water Storage Tank output valves. Additional soft spots of lesser significance are delineated in the final program report.[7]

The exemplary facility was selected as a convenient vehicle for demonstrating the methodology. It was not the purpose of this study to judge that facility. However, it was concluded that the facility is generally well protected against interactions considered within the scope of this study. A possible exception is the common failure of either pressurizer power operated relief valve and its isolation valve due to a shared location. This could lead to loss of the reactor coolant pressure boundary integrity. However, a more detailed study of the potential energy sources and environments in this location, and the failure modes of the components, would be necessary to determine the ultimate significance of this commonality.

ACKNOWLEDGEMENTS

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A VULNERABILITY ANALYSIS OF A PWR TO AN EXTERNAL EVENT

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ABSTRACT

The Vulnerability of a Nuclear Power Plant (NPP) to external events is affected by several factors such as:

(a) The degree of redundancy of the reactor systems, subsystems and components.
(b) The separation of systems provided in the general layout.
(c) The extent of the vulnerable area, i.e., the area which upon being affected by an external event will result in system failure. This area is dependent on the amount of protective shielding around it.
(d) The time required to repair or replace the systems, when allowed.

The present study offers a methodology, using Probabilistic Safety Analysis, to evaluate the relative importance of the above parameters in reducing the vulnerability of reactor safety systems. Several safety systems of typical PWR's are analyzed as examples. It was found that the degree of redundancy and physical separation of the systems has the most prominent effect on the vulnerability of the NPP.

INTRODUCTION

The vulnerability of a Nuclear Power Plant has been considered in the past from several different aspects. Chester and Chester [1] have analyzed a particular external event, that of a nuclear bomb attack. They found that under all probable conditions, the additional inventory of fission-products of the reactor core that could be released is insignificant, when compared with the damage inflicted by the nuclear bomb itself on the area surrounding the nuclear plant. Vulnerability of underground nuclear power plants to external events is considered differently [2,3]. In this case 10 to 40% of the total plant investment is spent to build it underground and provide the overall protection against external events. A different treatment of vulnerability can be found in the work by Okrent et al [9] on meteorites. Here the common understanding is that the probability that a meteorite will hit a plant is found to be very low ($<10^{-6}$ per reactor-year). Therefore, plants need not be designed to withstand meteorite impact.

European designed nuclear power plants [5] take into account several external events such as aeroplane crashes into the containment or industrial (chemical) explosions mainly due to transportation accidents in the vicinity of the plant. Against this eventuality the containment is hardened and a special emergency control room is added. The hardening provides overall protection against the external cause, and a separate remote control room, which includes all the required systems to perform shutdown and cooldown, provides protection by means of separation and redundancy.

In sabotage [6], it is assumed that the plant has been penetrated to a number of locations where systems vital to plant safe operation are damaged. Protection of the Nuclear Power Plant against sabotage is achieved by identifying the weak points and isolating them by several barriers difficult to overcome. The time required to penetrate these barriers is made sufficient so that security forces may arrive and take preventive action.

The present work deals with a different type of external event. This external event comprises striking the plant in a rather random pattern. Examples of this type of external event may be either on aeroplane or bomb hitting an Auxiliary Building which was not designed to withstand this occurrence. The occurrence is characterized by the striking of either a predetermined aiming point and Gaussian distribution of misses around the point or randomly distributed hits. In this work we study several parameters which influence the probability of loss of selected reactor safety systems, which in some cases could lead to a core meltdown accident. Some of parameters considered are separation of systems and subsystems, physical protection of vital areas and "heroic" repair of faulted components.

METHODOLOGY

Basic Assumptions used in the Calculations

The following assumptions have been used:

(a) Only a single external event damages the plant at any one time.

(b) The external event may cause damage to one or more systems. At the same time all other systems have their normal probability of failure.

(c) The normal probabilities of failures were taken from the Reactor Safety Study [7].

(d) The area affected by the external event is dependent on a large number of parameters. Some of which can be derived from experiments only. We used two different assumptions to scope their range:

  (1) A room which is struck by an external event is completely damaged, and all systems and equipment in that room fail to operate.

  (2) The room struck by an external event is damaged together with the neighboring room having a common wall (all systems and equipment located between the point of the external event striking and the second wall are destroyed and fail to operate).

  The floor is assumed to withstand the effect of the external event.

(e) Two values of accuracy in application of the external event are employed:

  (1) CEP (Circular error probability) = 20 meter. The choice of this value means that an external event of considerable accuracy has been intentionally aimed at a particular point.

  (2) CEP = 150 meters. This value means that a randomly distributed external event has struck the plant site. Its aiming point is immaterial and it can hit anywhere in the plant with almost uniform probability.

The General Method

The method employed in the present study includes the following steps:

(a) Study of how the systems operate and the general geographic layout in the power plant buildings.

(b) Construction of the fault-tree of the systems which include all events whose failure to operate causes system unavailability. The fault-tree is reduced to include only significant systems, subsystems and components which either:

  (1) have large failure rates or high probability of operator error.

  (2) are located in large rooms and have, therefore, relatively high probability of being struck by an external event.

(c) Assumption that an external event occurs in various geographical locations of the plant, followed by calculation of the fault-tree top-event unavailability for each case. This unavailability is designated as $P_{f,i}$:

$P_{f,i}$ = probability (System failure/external event at point i on the fault-tree)

(d) Assumption of the aiming point and a CEP. The aiming points are assumed to be:
   (1) Center of the Containment Building. j = 1
   (2) Center of the Auxiliary Building. j = 2
   (3) Center of the Condensate Storage Tank (CST). j = 3
   For each case the CEP can be either 20 or 150 meters.

(e) Calculation of the probability that one room is struck by the external event (only rooms containing equipment which are included on the FT are considered in this calculation). The result is $P_{c,i,j}$:

   $P_{c,i,j}$= the probability of an external event striking point i, when the aiming point is j.

   Pc,i,j is calculated taking the area of room i to be Si and its distance from the aiming point j to be R_{av} and the CEP chosen. Then,

$$P_{c,i,j} = \frac{0.5^{\left[\frac{R_{av}-\sqrt{S}/2}{CEP}\right]^2} - 0.5^{\left[\frac{\sqrt{S}/2+R_{av}}{CEP}\right]^2}}{2\Pi} \frac{\sqrt{S}}{R_{av}}$$

   Table I summarizes some values of R_{av}, S_i and $P_{c,i,j}$ for the several systems and equipment located in room i.

(f) Calculations of the total unavailability of the systems considered, as a result of an external event - $P_{T,j}$.

$$P_{T,j} = \sum_i P_{f,i,j} * P_{c,i,j}$$

Table I: Data used to calculate the probability of hit (Pc,i,j) of the vulnerable zones.

i	Vulnerable Zones	R_{av}(a) [meter] C A T	Area Si $[m^2]$	Common Wall with-room i	Probability of hit (Pc,i,j)			
					j = 1		j = 2	
					CEP=20	CEP=150	20	150
	Containment Bldg.							
1	Vital area A	8,24,120	160	-	0.074	0.002	0.032	0.002
2	Vital area B	8,40,136	160	-	0.074	0.002	0.006	0.001
	Auxiliary Bldg.							
3	D.C. area train A	45,18,100	420	5	0.009	0.004	0.118	0.004
4	D.C. area train B	45,18,90	420	5	0.009	0.004	0.118	0.004
5	Steam/F.W. pipe gallery	30,4,90	200	3 or 4 6 or 7	0.023	0.002	0.101	0.002
6	6.9KV train A	40,12,95	310	5	0.012	0.003	0.119	0.003
7	6.9KV train B	40,12,95	310	5	0.012	0.003	0.119	0.003
8	Comm. cable gallery	40,12,95	160	-	0.006	0.001	0.065	0.002
9	MDP-B room	50,22,75	26	10	0.0002	0.0002	0.006	0.003
10	TDP room	55,25,75	32	9	0.0001	0.0003	0.006	0.003
11	MDP-A room	40,18,95	130	-	0.005	0.001	0.039	0.001
	D-G Bldg.							
12	Train A	75,40,60	220	13	0(b)	0	0.008	0.002
13	Train B	75,45,60	220	12	0	0	0.004	0.002
14	CST	130,95,10	1250	-	0	0.007	0	0.009

(a) C=Distance measured from center of containment; A=Distance measured from center of Auxiliary Building; T=Distance measured from CST.

(b) negligible probablilty of hit, because of large distance.

Fault-Tree Evaluation

In the present study the PREP-KITT code [8] was employed for the evaluation of the fault-trees. KITT-2 was used because of its special feature of sequential "time-phases". This routine allows the change of the entire data set from one "time-phase" to another. A "time-phase" is a period of time during which the data is kept constant. This facilitates the inclusion of the external event in a simple way as follows:

(a) Time phase no.1: Calculation of system unavailability given no external events.
(b) Time phase no.2: System unavailability is calculated following the occurrence of an assumed external event.
(c) Time phase no.3: The same data is used as in time phsae no.1, except that system unavailability is calculated taking "heroic" repair rates.

Treatment of Common Cause Failures

Particular attention is required for the treatment of Common-Cause Failures (CCF) in the study of the vulnerability of Nuclear Power Plants to external events. An external event occurring in a certain room will generally have, a probability f1 of damaging a neighboring component (a), and at the same time, a smaller probability f2<f1 of damaging another component (b) at the other end of the room. There might also be a probability f3<f2<f1 of damaging components in the neighboring rooms. Thus, one needs to consider a CCF which can cause simultaneous failure of several components.

To include this kind of CCF in the fault-tree evaluations, an addition to the algorithm of the PREP-KITT code was required. This code treats all primary events on a fault-tree (FT) as being completely independent events. The CCF data of components i=1, i=2 or i=n, i.e. q_{12}, q_{13}, q_{1n} are associated (i.e. dependent) because the failure probability q_1 includes the CCF contribution of q_{12}, similarly the failure probability q_2 includes q_{21}, which is the same $q_{21} = q_{12}$.

$$q_i = \sum_{m=1}^{n} q_{im} - \sum_{m=2}^{n} \sum_{k=1}^{m-1} q_{im} q_{ik} + \sum_{m=3}^{n} \sum_{k=2}^{m-1} \sum_{l=1}^{k-1} q_{im} q_{ik} q_{il} + \ldots\ldots \ldots + (-1)^{n-1} q_{i1} q_{i2} \cdots q_{in}$$

$$i = 1,2,\ldots,n$$

where q_i = probability that event i occurs.

q_{ij} = probability of simultaneous occurrence of event i and j due to CCF.

q_{i+j} = probability of simultaneous occurrence of event i and j.

q_{ii} = probability of occurrence of event i alone (no occurrence of event j).

In the evaluation of the FT q_{i+j} has been used, and was included in the PREP-KITT calculations in the following way:

$$q_{i+j} = q_{ij} + q_{i'} q_{j'} - q_{ij} q_{i'} q_{j'}$$

$$q_{i'} = \sum_{\substack{m=1\\ m\neq j}}^{n} q_{im} - \sum_{\substack{m=2\\ m\neq j}}^{n} \sum_{\substack{k=1\\ m\neq j}}^{m-1} q_{im} q_{ik} + \sum_{\substack{m=3\\ m\neq j}}^{n} \sum_{\substack{k=2\\ k\neq j}}^{m-1} \sum_{\substack{l=1\\ l\neq j}}^{k-1} q_{im} q_{ik} q_{in} + \ldots\ldots$$

$$\ldots + (-1)^{n-1} q_{i1} q_{i2} \cdots q_{i,j-1} q_{i,j+1} \cdots q_{in}$$

$$i = 1,2, \ldots,n$$

To summarize this correction to PREP-KITT it has been shown that event (i) is the union of events (i') and (ij). The correct result from PREP-KITT is obtained when data on (i') is used as input rather than on (i).

APPLICATIONS

Two applications were made in the present study. The first application studied the Containment Spray Injection System (CSIS). The detail of this system were taken from the Reactor Safety Study [7]. However, the layout of the CSIS was taken from a specific PWR other than that of the RSS study. This application was particularly suitable to check the entire calculational method and to obtain some preliminary results. The second application studied the Auxiliary Feedwater System (AFWS). The system description and its layout were taken from one typical PWR.

Application 1: The CSIS

The CSIS was found to be distributed through four principal geographical zones in the Nuclear Power Plant:

(a) Zone no.1: The containment. Herein are located the spray nozzles in two redundant rings, and the check valves which provide containment isolation on the pipes leading to the spray nozzles.
(b) Zone no.2: The valve room. Herein are located most of the valves on the pipes leading from the pump room to the containment penetrations. In the layout studied there was a single valve room. A second case in which two separated valve rooms existed was also evaluated.
(c) Zone no.3: The pump rooms. Two pump-rooms were found in the layout.
(d) Zone no.4: The Refuelling Water Storage Tank (RWST). It was located outside the Auxiliary Building.

The aiming points used were as described in the previous section except that the RWST was considered instead of the CST. A table similer to table I was prepared [4] and the probabilities of hit $P_{c,i.j}$ were calculated.

Using the PREP-KITT code the probabilities of the CSIS unavailability as a function of time were calculated. Table II and Fig.1 summarize the results.

Table II: CSIS failure probabilities ($P_{f,i,j}$) before, at time of hit and after repair is completed.

Location of External - Event Hit	CSIS System Configurations and Locations		
	Pump Rooms	2	2
	Valve Rooms	1	2
	Separated Spray Nozzles in Cont.	1	1
	RWST	1	1
Pump Room	Before hit	0.0024	0.0024
	Time of hit	0.021	0.021
	After repair	0.0024	0.0024
Valve Room	Before hit	0.0024	0.0024
	Time of hit	1.0	0.021
	After repair	0.0024	0.0024
RWST	Before hit	0.0024	0.0024
	Time of hit	1.0	1.0
	After repair	0.0024	0.0024

The result for $P_{ci,j}$ and $P_{fi,j}$ were summed up as given in the equation for $P_{T,j}$. Table III summarizes the $P_{T,j}$ values for the CSIS.

Table III: Summary of the probability $P_{T,j}$ of CSIS failure as a result of an external event occurrence at point j.

External Event Aiming Points	CSIS System Configurations and Locations				
	Pump Rooms	2	2	2	2
	Valve Rooms	1	2	1	2
	Separated Spray Nozzles in Cont.	1	1	2	2
	RWST	1	1	1	1
Containment Building (j=1)	CEP = 20	0.235	0.230	0.013	0.007
	CEP =150	0.016	0.016	0.011	0.011
Auxiliary Building (j=2)	CEP = 20	0.185	0.182	0.125	0.122
	CEP =150	0.017	0.017	0.012	0.012
RWST (j=3)	CEP = 20	0.425	0.425	0.425	0.425
	CEP =150	0.017	0.017	0.012	0.012

It can be seen from this table that physical separation of systems can have a large effect in reducing the vulnerability of the plant. The area of the valve room is small and, therefore, the separation of the valves into two separated trains in two valve rooms has small impact. The area of the containment is quite large. Therefore, separation of the spray rings has a large impact. Fig 1 shows clearly the importance of fast repair efforts. FSUM (Fig 1) which is the cumulative probability of failure increases rapidly with time and approaches high values. The effect of the CEP is also significant and in the case of the RWST there is a factor of over 20 between the probabilities calculated for the two CEPs.

Application no.2: Auxiliary Feedwater System (AFWS).

The analysis of the AFWS included also the onsite electrical system (assuming the offsite power has failed simultaneously with the occurrence of the external event). Some failure modes of this system may eventually lead to core damage. It should be noted that this study of the AFWS considered only the first 8 hours of operation after the occurrence of an external event.

Table I summarizes the data used in the external event analysis. The probability of hit $P_{c,i,j}$ is shown for the vulnerable zones i considered and for the three aiming points j, i.e.. Containment (j=1), Auxiliary Building (j=2) and Condensate Storage Tank (j=3 , not shown). It can be seen that, in general, the probability of hitting a certain area is quite small for a large CEP, and may reach about 10%for accurate cases (CEP=20). The probability of striking the CST when it is assumed to be the target, is quite large (40%). However, the ESWS is a redundant system for water supply, therefore the effect on the overall AFWS failure probability is smaller than with other targets.

The fault tree of the various combinations of failures, which cause AFWS failure are shown in Fig.2. Each entry has a transfer-in from a particular subtree not shown in this paper."Failure" in all events (other than the top-event) means failure due to an external event. "Loss of" or "Out of operation" mean unavailability which may occur during normal operation. Each "and-gate" on the fault-tree combines an external event failure with failures in normal operation.

The results of this application are summarized in Table IV. It can be seen that "weak" common walls increase plant vulnerability significantly. For small CEP's and when the aiming point is the Aux. Bldg., AFWS failure becomes an almost certainty. The probability of failure of the AFWS is reduced to a few percent when common walls are hardened or rooms are separated geographically (i.e. "strong" common walls). Large CEP's result in most cases in relatively low probability.

Table IV: Summary of AFWS Failure Probabilities ($P_{T,j}$)

External Event Aiming Points	AFWS System Failure Probabilities - $P_{T,j}$			
	"Strong" Common Walls which withstand external event		"Weak" Common Walls, not limiting effect of ext.event	
	CEP = 20	CEP = 150	CEP = 20	CEP = 150
Containment Bldg.	8.1×10^{-3}	8.5×10^{-4}	9.8×10^{-2}	1.7×10^{-2}
Auxiliary Bldg.	4.0×10^{-2}	1.8×10^{-3}	8.3×10^{-1}	1.3×10^{-2}
CST	1.2×10^{-3}	8.4×10^{-4}	1.2×10^{-3}	1.4×10^{-2}

CONCLUSIONS

(a) The location of systems in large rooms increases vulnerability. Reduction of the area where a vital system is located, by means of protection or separation, could be effective in reducing probability of failure.

(b) Reduction of the vulnerable area by overall protection is less effective than physical separation of redundant trains. While separation can reduce the probability of failure by an order of magnitude, area protection by hardening will generally reduce the failure probability linearly proportional to the vulnerable area reduction.

(c) A small CEP is a critical factor. However, separation reduces the effect of the CEP on the failure probability. When the CEP is large the probability of system failure does not exceed a few percent, even if no special means are invested to improve separation or protection.

(d) When the external event is strong enough to penetrate neighboring rooms having common walls, than the probability of failure becomes close to unity. A common wall between related vital systems should be avoided if practical.

(e) The cumulative probability of system failure as a function of time increases and may approach unity after some time interval. Therefore, "heroic" repair could be important factor in reducing plant vulnerability to external events.

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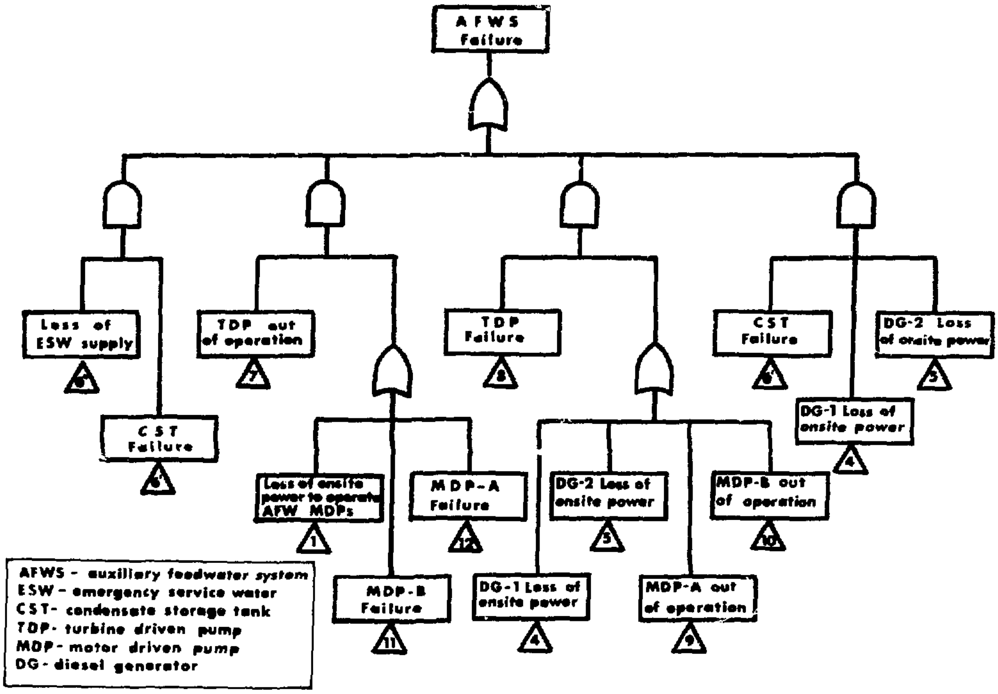

Fig. 2 - AFWS Fault Tree Combining External Event Failures and Normal Operation Failures.

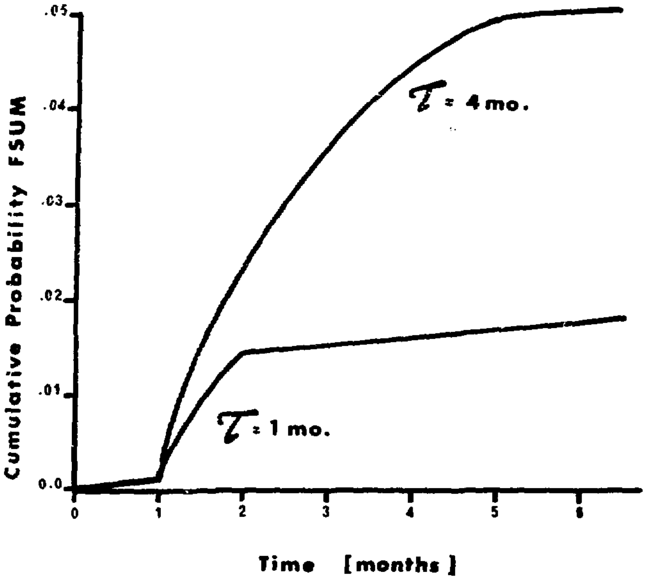

Fig. 1: The Cumulative Probability FSUM for at least one CSIS Failure upto Time T. External Event in one of two Pump or Valve Rooms.

Dup

A PATH TO DEVELOPMENT OF QUANTITATIVE SAFETY GOALS

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ABSTRACT

There is a growing interest in defining numerical safety goals for nuclear power plants as exemplified by an ACRS recommendation. This paper proposes a lower frequency limit of approximately 10^{-4}/reactor-year for design basis events. Below this frequency, down, to a small frequency such as 10^{-5}/reactor-year, safety margin can be provided by, say, site emergency plans. Accident sequences below 10^{-5} should not impact public safety, but it is prudent that safety research programs examine sequences with significant consequences. Once tentatively agreed upon, quantitative safety goals together with associated implementation tools would be factored into regulatory and design processes.

INTRODUCTION

There is a growing interest in defining numerical safety goals for nuclear power plants as exemplified by an ACRS recommendation [1] and a request by Chairman Hendrie to the ACRS [2] to come forward with a proposal. Numerical safety goals in some other applications are well advanced; for example, the goals for autopilot landing system reliability for aircraft at London Heathrow Airport, FAA criteria for airport traffic towers, HUD siting of housing projects in proximity of hazardous industrial complexes, etc., etc.

The General Atomic Company has been advocating use of Probabilistic Risk Assessment techniques in nuclear plant design, safety assessment and R&D guidance as well as in the regulatory process for several years [3], [4], [5], [6].

Major steps in the development and implementation of quantified safety goals are summarized in Figure 1. The figure suggests two avenues in deriving quantified safety goals. One avenue derives knowledge from comprehensive safety studies such as RSS [7], AIPA [8], and Deutsche Risikostudie [9]. These studies, plus subsequent improvements in uncertainty estimates, better understanding of consequences and explanations of meaning of low frequencies constitute sufficient basis for proposing interim, quantitative safety goals. The second avenue in deriving goals is by means of the risk budget concept.

INTERIM SAFETY GOALS

A great variety of accident sequences are examined when studying nuclear safety. An important aspect is the width of the spectrum of accident frequencies under consideration as shown in Figure 2. In the figure, the accident frequency or probability per unit time and per reactor is plotted against its consequences. Therefore, each accident sequence analysed can be shown as a point on the diagram.

There is a range of accident frequencies from, say, once per year down to somewhat less than once per hundred reactor-years (10^{-2}/year) where there exists an experience base for nuclear power plants. The limits on consequences, such as radiation doses, in this region are also fairly well established.

Some sequences of lower frequency must be accounted for in design so that more serious accidents which have a high probability of happening in the forthcoming years of the nuclear power program yield small and, thereby, hopefully acceptable consequences. Therefore, the design basis events must include accident sequences having frequencies below 10^{-2}/year.

The dividing line between the graphical region where design basis accident candidates are plotted and the lower regions where the sequences plotted are too unlikely to be DBA candidates is defined as the frequency limit line. The numerical value of this frequency was difficult to determine in the early years of nuclear power plant regulation because of limited development of the field of probabilistic risk assessment coupled with a lack of experience as to practicably achievable goals. In recent years, however, values for such a frequency have entered the regulatory process in some cases, and the capabilities for deriving such frequencies are rapidly improving. One derivation is given in Refs. 10 and 11, and a further step in the derivation process is given here.

A frequency of 10^{-6}/reactor-year has received widespread consideration as a future or ideal safety goal. The U.S. Atomic Energy Commission used this number in 1973 in a report on anticipated transients without scram (ATWS) (WASH-1270, Ref. 12) but did not examine whether it was practicable for nuclear power plants or reasonable compared to risks from other forms of electrical generation. The details of the definition of that frequency evolved over the years until in April 1978, in report NUREG-0460, Volume 1 [13], the frequency of 10-6/reactor-year was the goal for the requirement that ATWS not cause core melting or doses greater than 10CFR100 [14] values. In December 1978, in Volume 3 of NUREG-0460 [15], the frequency goal for ATWS was withdrawn. This action may facilitate a fresh look at what has been achieved in the licensing of nuclear power plants to date and how that insight influences practicable and reasonable goals.

Probabilistic risk assessments of some reactors have been made which can give some insight into practicable goals. Two that are important to consider here are the Reactor Safety Study and the HTGR AIPA study. (The German risk study [9], whose Appendices are not yet released, will also become important.) In these reports, the retrospective estimate for the median frequency for core melt in LWRs is 5 x 10^{-5}/reactor-year and the prospective estimate for core heatup in HTGRs is 3×10^{-5}. The uncertainty factors for these estimates (the number by which the median frequency is multiplied to obtain the upper bound frequency at the 95th percentile) are about 5 and 6, respectively. The estimate of the mean core melt frequency for LWRs is 8×10^{-5} based on the uncertainty factor of 5. The Risk Assessment Review Group concluded that the RSS uncertainties are greatly understated. In addition, the risk assessments described above were made prior to the accident at TMI and on only two specific plant designs. An examination of the TMI accident and the Reactor Safety Study shows that the TMI-type event was included in the RSS but raises the likelihood that probability values used in the RSS are not being achieved in all plants. This could lead to higher predictions of mean core melt frequencies for the totality of plants in the U.S. However, these accident frequencies implied by this line of reasoning are not considered acceptable; and, because of this, many improvements in safety are now being made in and for the power plants. Some of these improvements are in the direction to restore the former confidence that the frequency for severe accidents is low. It would appear to be possible to achieve a mean frequency for core melt as low as 8×10^{-5} if that was judged to be important. Since core melt is not a design basis accident, this frequency gives some insight about the limit line frequency for design basis accidents which is practicable.

Another consideration is of great importance in choosing the frequency limit line. Much effort, time, and money are spent on design features and associated research and development based on the choices of design basis events. We suggest that this effort is worthwhile if the chosen accidents are more likely to happen than not sometime in the entire nuclear power program of the United States. On the other hand, if it is more likely that the accident will not happen, it should not be chosen as a design basis event. For a national reactor program of 200 reactors, and assuming a 35 year lifetime for each, an accident with a mean frequency of 10^{-4}/reactor-year has about a 50% chance of happening.

Therefore, 10^{-4}/reactor-year is proposed as a future frequency limit line which forms a "design basis region" on Figure 2. The consequence limit proposed is that there be no identifiable public injury.

A "safety margin region" is needed below the design basis region to provide safety margin against some events whose probability of happening in the U.S.program is not very far below 50%. Events predicted to lie in the region of safety margin would not be expected to happen in the duration of a national reactor program, and so there should be no blanket requirement to automatically design for them thereby potentially unneccessarily increasing the cost of electricity generation. However, suitable margin against the unlikely chance that they may happen can be obtained by site emergency plans. If there are any cases where predicted consequences are particularly high, such as above some emergency reference level (ERL), then value/impact considerations are proposed to be used to consider choice of special design or operational options. This rationale for special provisions is only reasonable down to a certain small frequency suggested to be 10^{-5}/year.

The accident sequences below 10^{-5} should not impact public safety.

However, it is prudent that sequences with significant consequences be treated as candidates for a thorough safety research program with the ultimate objective of confirming a frequency below 10^{-5}/year. This paper, however, recommends a dismissal of accident sequences below 10^{-6} - 10^{-7} level, subject to collegial review (peer review) of low frequency, high consequence cases.

ACCOUNTING FOR UNCERTAINTIES

Uncertainties in frequency and consequence predictions can be significant and could influence the assignment of events to the design basis region. A calculation including uncertainties is needed of the probability of whether an accident is likely to happen or not in the U.S. program. Single values for frequencies are used though it is known that the predictions have uncertainty. The formula [10] for the single value of frequency, $\bar{\lambda}$, is

$$\bar{\lambda} = \iint \lambda \phi(\lambda, C) dC d\lambda$$

where ϕ is the joint probability density function describing the uncertainties in frequency, λ, and consequence, C. This single value yields the probability, P, that the event will happen in T reactor years of operation when used in the formula:

$$P = 1-\exp(\bar{\lambda}T)$$

The formula [10] for a single value of consequence, $\bar{C}$, including uncertainties, yielding the same risk (defined as the product λ and C) as all cases in the uncertainty distribution is

$$\bar{C} = \frac{1}{\bar{\lambda}} \iint C\lambda\phi(\lambda, C)\ d\lambda dC$$

RISK BUDGET CONCEPT

To foster a higher level of public understanding, nuclear risks must be presented in a better perspective by comparing them with risks associated with operation of, say, other means of electricity generation or chemical plants. The comparison should also include risks from other man-made hazards and natural phenomena, again, to balance public understanding. Technological assessment of risks is further supplemented by assessments of public perceptions. These several considerations could lead to a concept called "risk budget" shown on Figure 1. The definition of a long-term "risk budget" explicitly includes what is probably an inevitable constraint on nuclear power growth. The constraint is directly related to assessments of the public perceptions on the acceptability of nuclear power. As the public becomes more informed, the disparity between perceived risks and assessed risks diminishes, leading ultimately to a rational assessment of risks from all technological activities.

Since the range of technological activities is so broad and involves so many different organizations, there has been and will continue to be an institutional problem in adopting a risk budget. More adequate mechanisms are needed with which to resolve differences among groups. One might think that a Risk Council or Risk Department in the Federal Government would be a constructive step. The risk budget concept should be encouraged and developed.

APPLICATION OF QUANTITATIVE GOALS

There are several key activities for implementation and application of quantitative safety goals as seen in Figure 1.

Implementation requires continuing development in a number of areas. Methodology for fault tree and event tree analyses can be made more reproducible and economical. Consequence analysis can utilize improved assessments of uncertainty. The continued growth in the equipment failure data base is important. Value/impact methods could provide valuable insights for some problems. Supporting safety research can be more pertinent when guided by risk assessments. In some cases, options for safety enhancement may be needed. Common-cause failure development for redundant and diverse systems is of fundamental importance for highly reliable systems.

The overall methodology has more applications than just setting the top level of quantified safety goals, which are the frequency and consequence limits discussed in relation to Figure 2. The top-level goals are central to the problem of providing a basis for balanced and visible rules, but consider the additional applications shown on the right of Figure 1.

Some operating safety decisions are made prior to operation such as some of those in the technical specifications of plants. The combinations of equipment which are allowed to be out of service during operation should be chosen with overall safety goals in mind.

Decisions during operation could be improved by using computer modeling to check the change in risk in a plant during operation when components are taken out of service. One conceptual computer model for this function being developed at General Atomic is called the Predictive Incident Evaluator (PIE).

Quantitative safety criteria as proposed herein are compatible with the selection of design basis accidents (DBAs) for Safety Analysis Reports. This approach to design bases for safety allows the development of deterministic licensing criteria for DBAs.

Design criteria can also then benefit in conjunction with deterministic licensing criteria. This should improve the correlation between a suitably low safety risk and a suitably low investment risk by reducing the chances of radioactivity release within the plant. This might be especially achieved by using the precursor events which happen in plants as input data to the risk assessments.

Advanced concepts can be examined by prospective studies to systematically show where strong or weak points are in a design. This can provide valuable guidance in developing the safety characteristics of the concept.

There are, therefore, many potential rewards in moving to the use of quantitative safety goals. It is believed that continued development will reduce or solve the difficulties. The field should be given our wholehearted support.

ACKNOWLEDGEMENT

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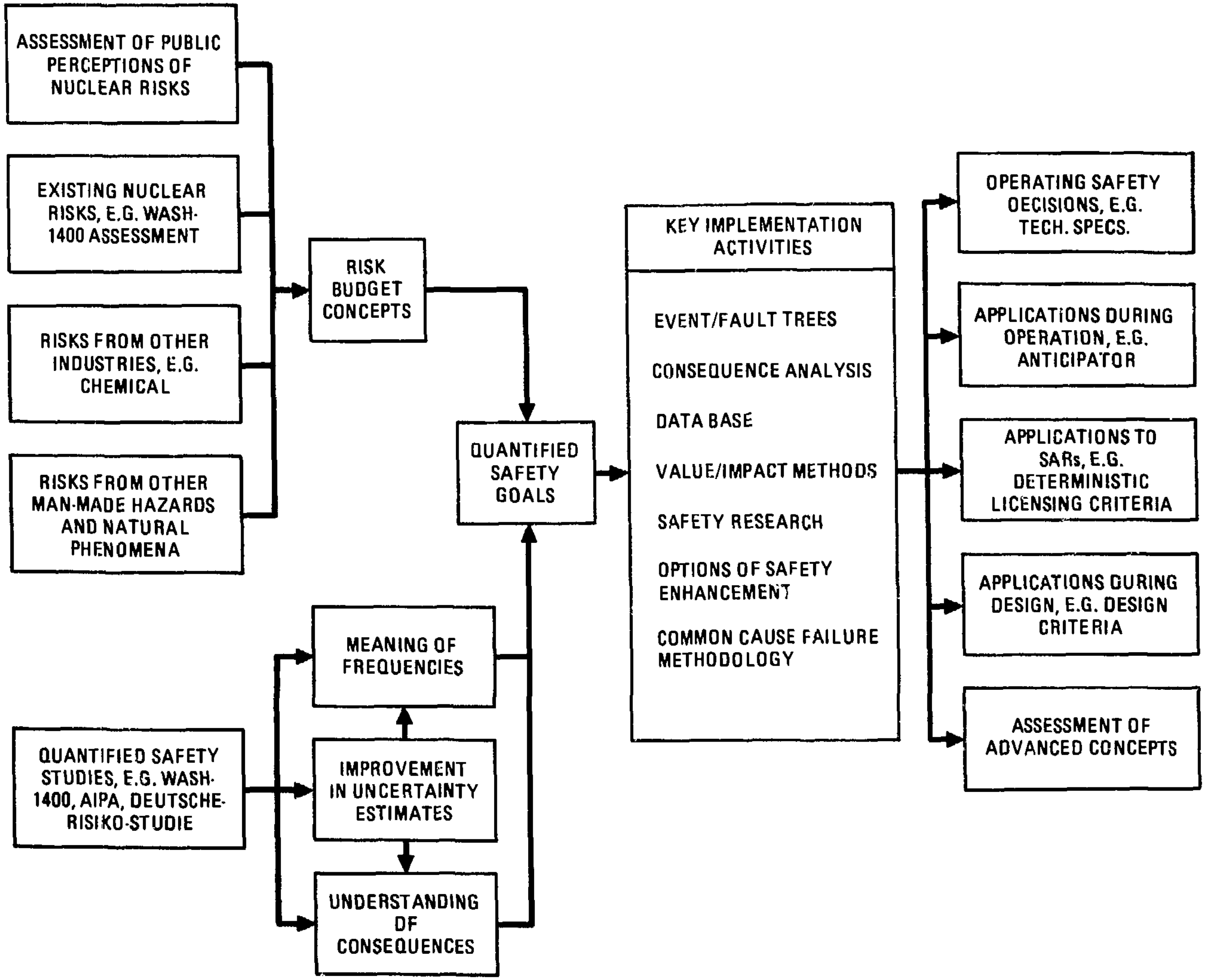

Figure 1. Major steps in development and implementation of quantified safety goals

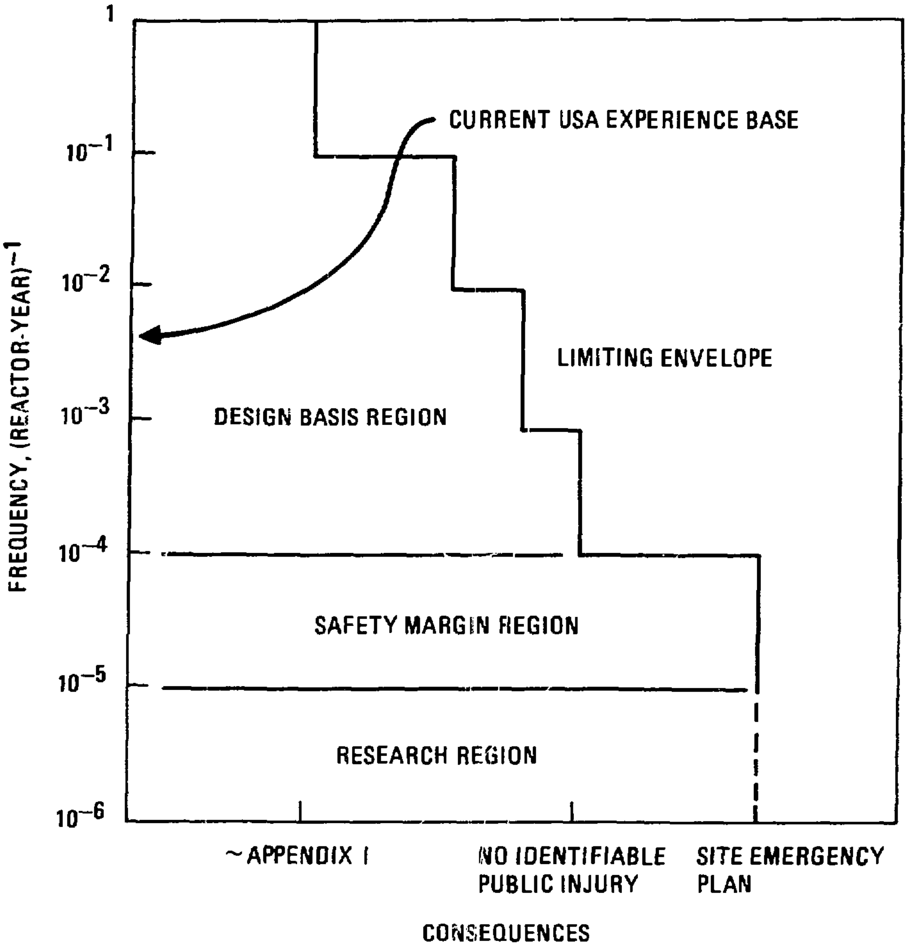

Figure 2. Interim map of quantified safety regions

Dup

MONTE CARLO METHOD FOR UNCERTAINTY ANALYSIS OF HTGR ACCIDENT CONSEQUENCES

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ABSTRACT

A method for propagating the uncertainties in the prediction of postulated accident consequences was devised. The method uses simplified mathematical models, derived from first principles and detailed deterministic computer model results to describe controlling physical processes in the transport of radioactive material from the plant and resulting population doses and health effects. These simplified models are used in a computationally efficient Monte Carlo error propagation technique. The approach is shown to be useful for uncertainty analysis of core heatup accidents postulated for the High Temperature Gas-Cooled Reactor. In combination with accident frequency predictions, the method provides the complementary cumulative risk curves typically used to present the final results of a risk assessment study.

INTRODUCTION

One aspect of probabilistic risk assessment methodology for nuclear power plants that has been attracting recent attention is uncertainty analysis of accident consequences [1-4]. The problem considered here is to find the probability distributions that describe the uncertainty in estimates of one or more accident consequence variables which are functions of many independent variables whose uncertainty can also be represented by assigning probability distributions. The availability of deterministic methods, such as mechanistic accident analysis computer programs, are assumed for point estimate predictions of the consequences. However, these deterministic methods do not yield per se simple functional relationships between the consequence variables and input variables in an analytical form suitable for statistical analysis, such as Monte Carlo simulation. Long computer running time makes impractical the direct use of the computer programs in a Monte Carlo simulation.

One possible solution for this problem is the use of surface response techniques [1,2]. However, it may be economically prohibitive to generate sufficient detailed point estimates with the deterministic computer programs to accurately fit a non-linear response surface. This applies particularly where multiple accident sequences and many input variables are involved.

Since the point estimates required to fit a response surface to one consequence output variable may not be suitable for another consequence output variable, additional detailed point estimates are often required. Also, since the response surface equations may not be based on physical considerations, the tails of the resultant output distributions may be invalid. Thus, a more direct method was developed and applied as described in this paper where the fundamental models are simplified to the extent that thousands of Monte Carlo trials can be run economically.

The primary advantages for the use of such an uncertainty analysis in probabilistic risk assessment are that it provides:

- a means of determining the uncertainty in the prediction of accident consequences in a consistent fashion with the methods quantifying uncertainty in predicted accident frequencies;
- a quantitative probability statement which replaces the qualitative notions of conservatism, realism, and pessimism;
- a method for carrying out sensitivity studies in support of design optimization for safety and guidance for safety research.

In this paper, the unique features of the suggested method are described followed by a discussion of its application to HTGR accident consequences.

METHOD

The goal of the current method is to derive simultaneous probability distributions of one or more accident consequence or output variables. The consequences are dependent on a common set of independent variables. The method consists of five steps:

1. Perform realistic point estimate accident calculations using detailed deterministic models.
2. Select the appropriate consequences or output variables, C_i, for which probability distributions are desired. For example, the consequence variables may be organ doses, curies released to the environment or, public health effects.
3. Construct simplified consequence models for the controlling physical phenomena identified in Step 1. The functional dependence on the independent variables, V_n , may differ for the various C_i., $C_i = f_i (V_1, V_2, \dots V_n)$.
4. Develop probability distributions for the independent variables on which the simplified models are based. These can be specified parametrically or in tables of percentiles.

5. Perform Monte Carlo samplings of the independent variable distributions developed in Step 4 to determine the uncertainty distributions of the consequence variables, C_i, as calculated by the simplified models.

Step 3 encompases the unique aspects of this scheme. Whereas, in principle, the functions f_i could be detailed deterministic models used to generate point estimates of accident consequences, computer expense generally prohibits this approach. The present approach recognizes that the deterministic models and results can be examined to extract information on the controlling physical phenomena and mathematical form of the output. This is unlike response surface methods which employ non-linear multivariate regression analysis in place of the deterministic models. The greater transfer of information from the detailed models in the current method means that fewer deterministic point estimates are required than in response surface methods. Moreover, point estimates of just portions of the detailed calculations can be compared with intermediate results of the simplified models. This further reduces the requirements for detailed calculations in order to construct the functions f_i. Thus, the unique nature of the present method is the use of closed form equations derived from physical laws, which approximate the detailed model output.

Consequence assessments of nuclear power plants involve the simulation of time-dependent transport and decay of radionuclides through plant barriers and subsequent environmental dose response. For a time interval t_o to t during a postulated accident sequence, the integrated activity release is given by

$$Q_a = \gamma_c \gamma_p q_{p,o} \left\{ \frac{1}{A} \left[1-e^{-A(t-t_o)} \right] - \frac{1}{B} \left[1-e^{-B(t-t_o)} \right] \right\} + \frac{\gamma_c q_{c,o}}{B} \left[1-e^{-B(t-t_o)} \right] \quad (1)$$

where

γ_c, γ_p are the leak rates from the containment and primary coolant boundary, respectively

$A = \lambda_d + \gamma_p$ and $B = \lambda_d + \lambda_c + \gamma_c$ in which λ_d, λ_c are the radioactive decay and containment cleanup rate, respectively

$q_{p,o}, q_{c,o}$ are the activities in the primary coolant and containment at time t_o.

The foregoing equation expresses the fission product release of a single nuclide to the atmosphere in a time period in which the variables such as removal rates (λ) and leak rates (γ) remain constant. As these variables may be time-dependent due to physical phenomena occurring during the course of the postulated accident, the integration in practice is broken up into time intervals over which the leakage and removal rates may be taken as constants.

If j is a subscript denoting each of the radionuclides considered and k is a subscript denoting each of the time intervals, the total integrated release to the atmosphere for each nuclide is $\sum_k (Q_{a,j})_k$. Personnel exposure doses D_i are then given by

$$D_i = C_i \sum_j \sum_k (Q_{a,j}) E_{j,i} \tag{2}$$

where C_i is related to atmospheric dispersion and breathing rate, and $E_{j,i}$ is the dose /curie conversion factor for nuclide j.

The consequence variables defined under Step 2 can be the organ doses D_i, the curies released $Q_{a,j}$ or other measures of accident consequences deemed appropriate. Equations 1 and 2 illustrate the forms of possible consequence variables C_i used for Step 3. The solution equations are programmed into a computer subroutine. A computer program named STADIC [5] is employed to perform the Monte Carlo simulation described in Step 5 above. Cumulative probability distributions of the defined independent variables are specified as input to the program. The variables to which the consequence outputs are insensitive are in effect treated as constants. The STADIC program provides for the user-supplied FORTRAN subroutine representing the simplified consequence models to determine the C_i. When the desired number of samples is obtained, the output routine calculates percentiles of the uncertainty distribution of each consequence variable C_i along with the distribution mean and the standard deviation. Since the dependent variables, C_i, are calculated simultaneously in STADIC, correlations in their uncertainty distributions are automatically accounted for.

APPLICATION TO HTGR RISK ASSESSMENT

The Monte Carlo method described above has been applied to uncertainty analyses of High Temperature Gas-Cooled Reactor (HTGR) accident consequences [6,7]. Illustrated here is the analysis of HTGR core heatup accidents, which are initiated by a postulated loss of forced primary coolant circulation. In the HTGR these events are characterized by long thermal response times, on the order of days, due to the large heat capacity of the graphite core and prestressed concrete reactor vessel (PCRV). Simplified consequence models $f_i(V_1,V_2, \ldots V_n)$, were derived for five dominate core heatup scenarios. These were considered representative of the more than 50 core heatup sequences, with frequencies above 10^{-9}/reactor-year, identified in a comprehensive probabilistic risk assessment for HTGRs [6].

The deterministic computer programs indicated that the radiological doses and health effects of postulated HTGR core heatup accidents is controlled by just twelve radionuclides. The physical phenomena governing the release of these radionuclides was found to be scenario dependent. When the containment remains intact throughout the accident, the release is governed by the containment leak rate and transport time allowing decay of noble gases released to the containment. In core heatup scenarios with containment failure, the release is controlled by (1) the time of containment failure and its subsequent leak rate; (2) the amount of radionuclide release from the PCRV after containment failure, if any; and (3) the extent to which condensible nuclides are removed from the containment atmosphere due to filtered removal, plateout, fallout, etc. Figure 1 depicts the phenomena and variables and interrelationship considered by the system of equations developed for the consequence functions in the STADIC code subroutine. Fifteen independent variables were identified as being adequate and probability distributions were assigned to each on its own merit, including parametric and non-parametric distributions. The time intervals in which removal constants and leak rates were kept constant were defined by the radionuclide core release time, start, or failure time of the containment, and an arbitrary endpoint (thirty days).

Three inhalation doses (thyroid, lung, and bone) along with whole body gamma external exposure were evaluated at 2.5 km for a representative U.S. site using the 12 nuclides of importance. Since the magnitude of organ doses encountered fall in the range where no acute illnesses or fatalities are likely, these were combined based on their relative contributions to latent health effects. The combined dose, termed "health effects dose, D_{He}" is such that 1 health effects rem equals 0.13 latent cancer fatalities and is appropriate for latent health effects only.

Figure 2 compares the probability distributions of an intermediate consequence output variable (C_1 = containment failure time), for the relevant core heatup scenarios as predicted by STADIC. During a postulated core heatup, the HTGR containment may fail as a result of the burning of flammable gases produced as the PCRV concrete is decomposed. Figure 3 presents the conditional complimentary distribution of organ doses given the occurrence of an HTGR core heatup with containment failure by gas accumulation.

Each core heatup sequence with a frequency greater than 10^{-9}/reactor-year can be assigned to a release category represented by one of the dominant scenarios. By combining the cumulative probability distributions with the release category occurrence frequencies, release category risk curves are derived. The overall HTGR core heatup risk assessment curve is presented in Figure 4.

CONCLUSION

Primary advantages of the presented method appear to be flexibility and economy. Experience has shown that a much better understanding and cross check of the physical phenomena is afforded by use of the method in concert with the detailed deterministic computer programs. Application to HTGR accident consequences has facilitated a clearer perception of risk by quantifying risk envelopes of dominant accident sequences and combining these in the overall risk envelope [6]. Use has been made to evaluate the effect of containment design options on the overall risk envelope [7]. Future use to quantify the relative influence of input variables for HTGR safety research guidance is anticipated.

ACKNOWLEDGEMENT

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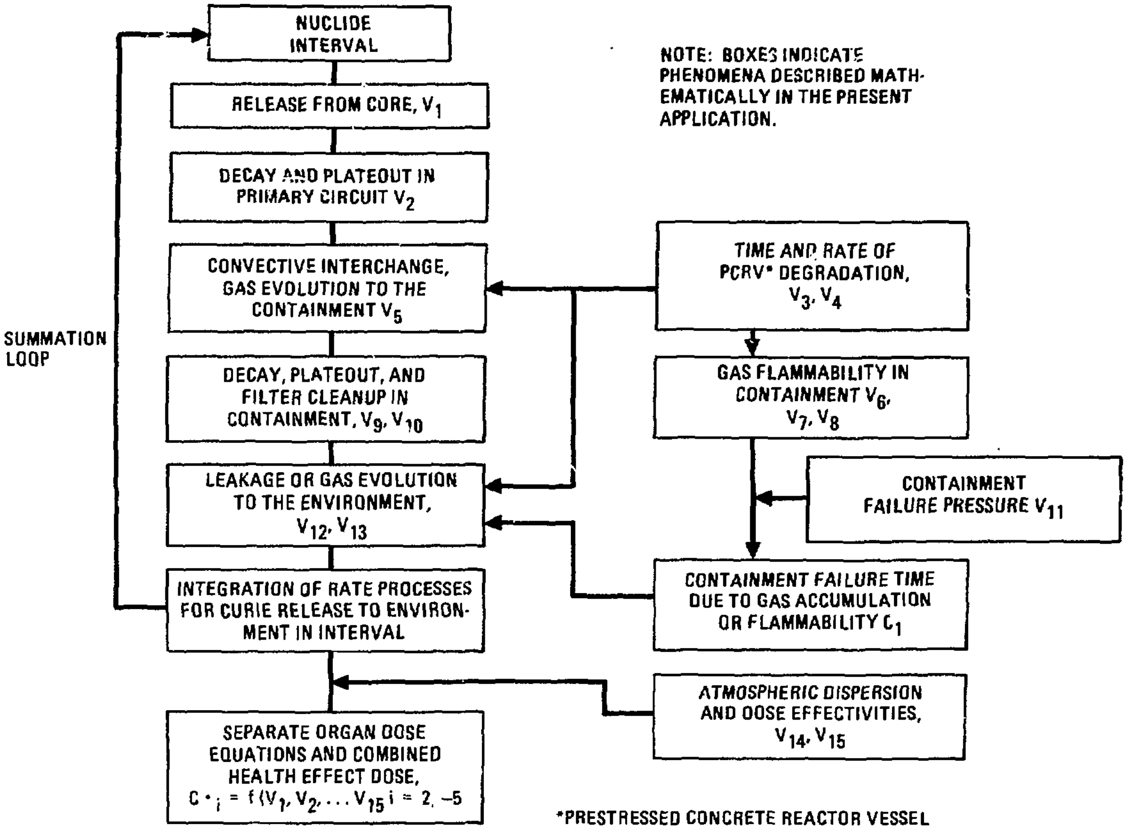

Fig. 1. Calculation of the consequence functions, C_i, for HTGR core heatup sequences

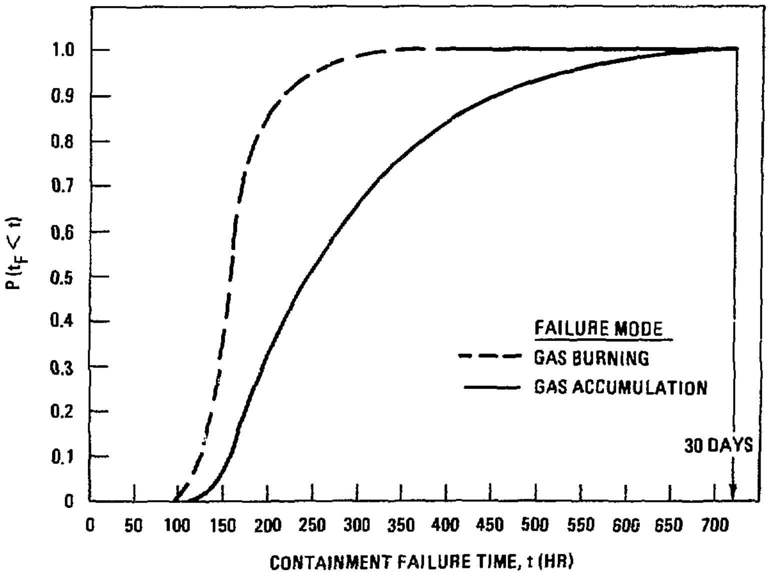

Fig. 2. Cumulative probability distribution function of containment failure time for unrestricted core heatup scenarios

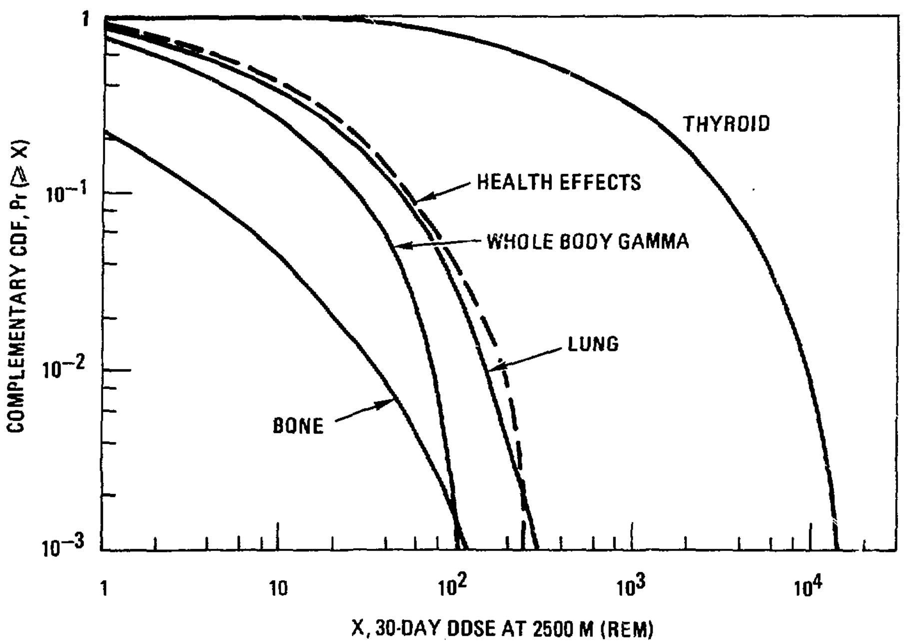

Fig. 3. Complementary cumulative distribution of doses at 2.5Km given occurrence of core heatup with containment failure by gas accumulation

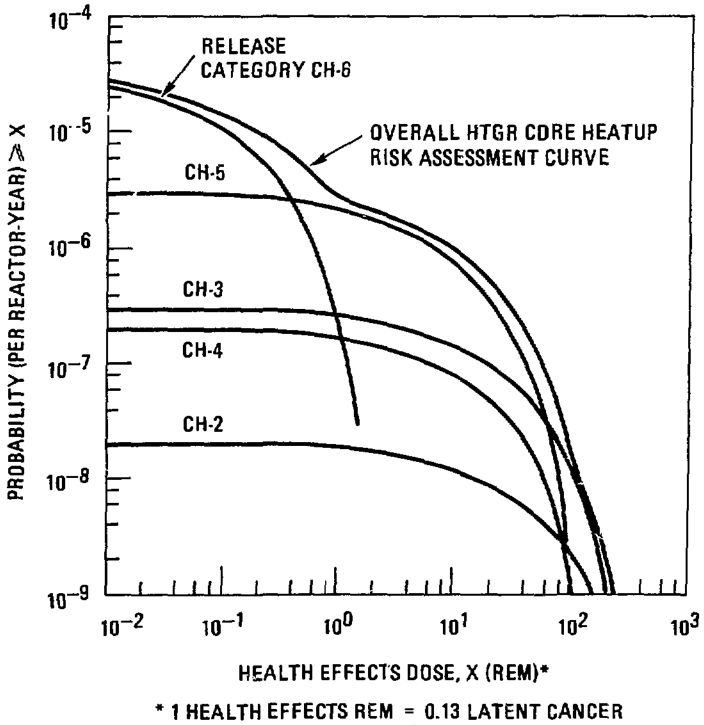

Fig. 4. Release category complementary cumulative curves of health effects dose (CH-1 not shown due to extremely low probability)

EVOLUTION OF DESIGN REQUIREMENTS TO ACCOMMODATE CLASS 9 ACCIDENTS DURING FLOATING NUCLEAR PLANT LICENSING REVIEW

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ABSTRACT

As part of the Final Environmental Statement for the application for a license to manufacture Floating Nuclear Plants (FNP), NRC required that the concrete biological shield beneath the reactor vessel in the FNP design be replaced with a layer of refractory magnesium oxide or equivalent material to provide increased resistance to melt-through in the event of a postulated core-melt accident. It was further concluded that future applicants for siting an FNP at an estuarine or riverine site must provide an essentially impermeable basin enclosure so as to limit introduction of radioactivity into the surrounding water body in the event of a postulated core-melt accident. The requirements resulted from an on-going series of studies of dose consequences and risk resulting from release of core debris and contaminated containment sump liquids to the water surrounding an FNP in the event of a core-melt accident (Generic Liquid Pathways Studies). This paper traces the evolution of the environmental regulatory requirement to add design features to mitigate the consequences of a core-melt accident over six-years of licensing reviews and attempts to identify some of the factors leading to its imposition.

INTRODUCTION

In Part III of the Final Environmental Statement [1] related to the manufacture of Floating Nuclear Plants (FNP), the NRC required that the four-foot-thick concrete biological shield mat beneath the reactor vessel be replaced with a layer of magnesium oxide or equivalent refractory material that would provide increased resistance to melt through by a molten reactor core for a postulated core-melt accident. The refractory material was to be such as to not react with core-melt debris to form a large volume of gases. In addition, NRC required that future applicants applying for a license to locate an FNP at an estuarine or riverine site must provide an essentially impermeable enclosure so as to limit introduction of radioactivity into the surrounding water body in the event of a core-melt accident. These two design requirements were specified by NRC for the purpose of protection of the enivronment. They represent the first imposition of design requirements for mitigating the consequences of core-melt accidents on light water reactors since a preliminary design of a "core-catcher" was requested during the Construction Permit review of the Indian Point-2 unit[2]. During Indian Point-2 detailed design, it became apparent that other engineered safety features already incorporated in the design would provide a high degree of protection against core-melt and that the "core-catcher" could not be demonstrated to provide the hoped for benefits. The device was removed from the Indian Point-2 design.

This paper traces the evolution of the regulatory environmental design requirements for the FNP over six years of FNP licensing review and attempts to identify some of the factors responsible for their imposition. A recent NRC Commissioner's decision upheld the NRC Staff in their consideration of core-melt accidents in the FNP environmental review. Implications of this Commission decision for land based plants are also briefly addressed.

HISTORY

The FNP licensing review process has been arduous and drawn out although the FNP represents little departure from conventional design. The FNP utilizes a 3427 MWt Westinghouse PWR Nuclear Steam Supply System adapted for mounting on a 400 foot by 378 foot floating platform. The Nuclear Steam Supply System is relatively standard with most of the design changes being in the balance of plant area. Many of the balance of plant design differences resulted from an NRC requirement that the plant have capability for being placed and maintained in a safe shutdown condition in the event of non-mechanistic sinking emergency (i.e., plant bottomed and flooded)[3]. The FNP design employs an ice condenser containment.

With respect to licensing requirements, Appendix M to 10CFR50 was promulgated in 1973 [4] and set forth licensing requirements for a standardized plant, such as an FNP, which was to be manufactured at a location different than its eventual operating site. Among other things, Appendix M requires (1) a set of site parameters related to the design be set forth in the plant design (safety) report (these parameters were denoted as plant-site interface parameters by Offshore Power Systems (OPS)), and (2) that an environmental report be prepared directed at the construction and operation of the reactor at sites having characteristics which fall within the envelope of the postulated site parameters.

The FNP design concept was submitted to the then Atomic Energy Commission for a pre-application review to determine if there were factors which would preclude licensing of the FNP (1971-72). A review by ACRS at this stage was also requested. The Regulatory Staff did not identify any factors which would preclude licensing, although the Staff did require that the plant be designed to permit safe shutdown in the event of a postulated non-mechanistic sinking of the platform [3].

Questions were raised by the ACRS regarding possible core-melt accidents in both the Offshore Power Systems pre-application review in 1972 and the immediately following pre-application review of the Atlantic Generating Station offshore FNP site in 1973. The 1972 ACRS letter [5] stated that further consideration should be given to possible means for assuring maintenance of containment integrity in the highly unlikely event of core melt-through, considering the possible advantages of the readily available source of water.

The 1973 ACRS letter on the Atlantic Generating Station site [6] recommended that the possible advantages to safety of a closed breakwater be evaluated and particularly the effectiveness of a closed breakwater for mitigating the possible consequences of a very low probability uncontained fuel-melting accident. The letter also noted that further work is needed on the dispersal characteristics of fission products and plutonium which might be released to the surrounding water body.

The issues raised by ACRS were investigated by OPS and two reports were submitted to the Regulatory Staff in late 1974. One reviewed existing technology respecting molten core retention devices and the use of such devices for maintaining containment integrity. The report [7] concluded that the presence of a large heat sink represented by the water surrounding the FNP is not, by itself, a particular advantage. Rather, ability to arrest a molten core is determined by heat removal from within the mass of core and structural debris. The second report [8] contained preliminary evaluations of dose consequences that could result from introduction of core-melt debris into the ocean. The report concluded that, with curtailment of access to nearby beaches and the curtailment of public consumption of contaminated fishes, individual and population doses via liquid pathways from a postulated core-melt accident would be acceptably small for ocean sited FNPs. The contents of these reports were reviewed with ACRS in early 1975. ACRS requested at that time a detailed evaluation of the studies by the NRC Staff before they were willing to reach any findings.

The ACRS action added a new dimension to core-melt accident considerations for the FNP. Until this time NRC had been passive, generally taking the approach that consideration of core-melt accidents was not required by the regulations. The recognized difference in the environment which the debris would experience following melt-through (water rather than soil or rock) had not been considered significant enough by the Staff to require additional studies.

LIQUID PATHWAYS STUDIES

After six months, the NRC Staff initiated action. OPS was required, as a condition for continuing the license application review, to participate with NRC in a joint study of post-accident dose effects via liquid pathways. Initially, the study was to focus on dose effects rather than core-melt accident scenarios. In fact, the study scope was described by the NRC as evaluating the consequences of "dumping a bushel basket of fission products into the basin surrounding an FNP". The calculated consequences were then to be compared with those for a land-based plant for similar releases of radioactivity. As the study progressed, it was decided that generic types of accidents would be utilized as a basis for estimating quantities of radioactivity released to liquid pathways. Generally, the classes of accidents contained in the proposed annex to Appendix D of 10CFR50 [9] were utilized for accidents within the design basis. For the core-melt accident, source terms associated with release of contaminated containment sump liquids and with the leaching of core-melt debris by basin water were considered. Core-melt accident scenarios were not considered in detail at this stage.

Reports for the first series of Liquid Pathways Generic Studies (LPGS) were issued by OPS [10] and NRC [11] in 1976. The NRC report was referenced in Draft Part III of the Environmental Statement[11]. The NRC LPGS report concluded that consequences via liquid pathways for FNPs would be comparable to and within the consequences assessed for land-based plants. DES-III concluded that the consequences of releases to liquid pathways were not significant when compared with potential airborne releases.

During review of the reports by ACRS, there was substantial questioning regarding the adequacy of the source term assumptions employed for the study, particularly on the subjects of the accident mechanistics associated with various accident scenarios and the effect of the scenario assumptions on releases to the liquid pathways. A substantial number of questions also resulted from the formal environmental review of DES-III. Perhaps most significant were questions related to core-melt accident mechanistics, their effect on liquid pathways source terms, and requests from other government agencies that dose consequences for riverine and estuarine sites be evaluated in more detail.

As a result of these comments, NRC unilaterally chose to significantly expand and modify the liquid pathways generic study particularly in the areas already cited. A new series of greatly expanded LPGS reports were issued in late 1977 by OPS [12] and early 1978 by NRC [13]. Generally dose consequences via liquid pathways (both individual and population doses) were not substantially different than those reported earlier for the ocean sites. However, dose consequences for land-based plants via liquid pathways were calculated by NRC to be substantially lower than reported previously by NRC, while for FNPs at estuarine sites dose consequences were calculated by NRC to be higher than the consequences for FNPs at ocean sites. More detail regarding the results of the OPS Liquid Pathways Study are presented elsewhere [14].

NRC IMPOSITION OF ENVIRONMENTAL DESIGN CONDITIONS

The NRC assessment of the significance of the results in the second set of LPGS reports was contained in the Final Environmental Statement, Part III (FES-III), a licensing approach to which OPS objected. NRC concluded that liquid pathways consequences from postulated core-melt accidents were significantly greater for FNPs than land-based plants. The NRC also concluded that liquid pathways releases at FNP estuarine sites might lead to long term contamination of the estuarine environment. In reaching this conclusion, conservative assumptions were employed in the NRC cost-benefit evaluations since adequate models did not exist to describe the impact on estuaries. It is apparent from FES-III that the Staff findings for estuarine sites is based on perceived socio-economic impacts via liquid pathways rather than estimates of impacts based upon calculations or data.

The NRC also compared calculated liquid pathways core melt consequences to those calculated for air pathways for core-melt accidents. NRC concluded that dose consequences via liquid pathways (which are not very different for any accident scenario assuming melt-through to liquid pathways has occurred) were comparable to dose consequences via air pathways for the "more likely core melt scenarios" out of the total class of "highly unlikely core-melt accidents". OPS contended that the "more likely" of the core-melt accident scenarios were minor contributors to total residual risk via air pathways for core-melt accidents. Further, individual doses via liquid pathways are small and do not lead to predicted fatalities as is the case for air pathways. OPS therefore concluded that dose consequences via liquid pathways were not significant when compared to dose consequences via air pathways for core-melt accidents.

Based on the NRC conclusion cited above, NRC required in the FES-III[1] that estuarine and riverine site features would be required for an FNP to limit to low levels the radioactivity that would be released to the open water body in the event of a postulated core-melt accident (for protection of man and the ecosystem). Another requirement, applying to the FNP design, was a device for delaying melt-through of molten core debris. The NRC purpose for adding this device was to provide time for initiation of interdiction activities prior to melt-through, should a highly unlikely core-melt accident occur.

With respect to consideration of core-melt accidents in the FNP environmental review, OPS took the position that the proposed Annex to Appendix D of 10CFR50 [1] ,as well as subsequent court cases, precluded consideration of Class 9 accidents. For example, the proposed Annex, which had been the basis for a number of court decisions excluding the need for Class 9 accident considerations in environmental reviews, states that Class 9 events involve sequences of postulated successive failures more severe than those utilized in establishing design basis events. While these consequences may be severe, their probability of occurrence is so small that their environmental risk is extremely low. For these reasons it is not necessary to discuss such events in an applicant's environmental report.

There was a general agreement between OPS and the Staff that the probability of occurrences of Class 9 events for an FNP and land-based plants were substantially the same. Staff consideration in the FNP environmental review was therefore based on potential differences in consequences.

HEARING BOARD APPEAL

OPS, via the Licensing Board appeal process, sought to have voided the NRC Staff consideration of core-melt events in the FNP environmental review. The basis for this appeal was past NRC practice which we believed precluded such consideration in the environmental review process.

The Appeal Board, by a 2-1 split decision, ruled that consideration by the Staff of Class 9 events in the FNP application was acceptable [15]. The basis for the Appeals Board finding was the proposition that, while the probability of a core-melt accident may not be greater or its consequences more severe for an FNP, risks may be of a different kind than those associated with plants sited on land. Thus, since risks for an FNP may be different in kind from those which formed the basis for the regulatory guidance in the proposed Annex, consideration of core melt accidents was judged to be appropriate. The Appeals Board did hold in its decision (contrary to the NRC Staff position) that the proposed Annex and past Appeals Board decisions had excluded Class 9 accidents from consideration in Environmental Statements solely on the basis of their small probability of occurrence without consideration of consequences. The Appeals Board further held that the wording of the Annex that indicated Class 9 events "need not be considered" is properly read as an exclusion of their consideration. Thus the Appeals Board's decision did not alter existing practice with respect to consideration of Class 9 or core-melt events in licensing proceedings for land-based plants unless perhaps it could be demonstrated that consequences were different in kind from those of previously licensed land-based plants. The Appeals Board also noted that once the NRC Staff LPGS study was completed, inclusion of the study in the Environmental Statement was necessary under the full disclosure provisions of NEPA.

COMMISSION APPEAL

The Appeals Board decision was appealed and certified by the Appeals Board to the Commission for their consideration. The Commissioner's upheld the Appeals Board majority decision although on a somewhat different basis. The Commission invoked their policy-making prerogative in finding that consideration of Class 9 events in environmental reviews was appropriate if risks were different in kind from Class 9 accident risks at land-based reactors. The full disclosure aspects of the National Environmental Policy Act were also stressed as an additional reason for considering the Class 9 accidents in the FNP environmental review.

In its decision the Commission noted they were not expressing any views on environmental consideration of Class 9 accident at land-based reactors. They did note concern with this question and their intent to pursue a rulemaking to re-examine Commission policy in this area. In this decision the Commission asked the NRC Staff to:

1. Provide its recommendations on how the interim guidance of the Annex might be modified on an interim basis, and until the rulemaking on this subject is completed to reflect developments since 1971, and to accord more fully with current Staff policy in this area; and
2. In the interim, pending completion of the rulemaking on this subject, bring to their attention any individual cases in which it believes the environmental consequences of Class 9 accidents should be considered.

These two requests make it clear that NRC policy respecting Class 9 and core-melt accidents are being re-examined and may change.

IMPLICATIONS AND CONCLUSIONS

Since the Commission's decision with respect to consideration of Class 9 and core-melt events in the FNP environmental review, the pace of regulatory recommendations regarding consideration of Class 9 events has quickened. Certainly the TMI-2 event has been a contributing factor. Specifically, the Siting Policy Task Force of NRC recommended in their report [17] that the population criteria of 10CFR100 be revised, that emergency planning for severe accidents be required and that measures be required to mitigate the effects of releases of radioactivity to liquid pathways in the vicinity of a site following a core-melt accident. As part of the TMI-2 Lessons-Learned Task Force final report [18], the Task Force recommended a notice to conduct a rulemaking which would solicit comments on the need for design features to mitigate consequences of accidents involving either core-melt or severe core damage. All indications are that re-examination and possibly revision of regulations related to consideration of Class 9 and core-melt events are likely for land-based plants similar to what has already occurred for FNPs.

Consideration of core-melt accidents during the FNP licensing review has been an evolving process over the past six years. These considerations culminated in the requirements set forth in the OPS environmental review documents (FES-III) specifying (1) that a device, for delaying melt-through of the core for a minimum of two days, be included in the FNP design and (2) that the enclosure about FNPs sited at riverine or estuarine sites be designed so it can be quickly made relatively leak-tight to limit to low levels the release of radioactivity to the surrounding water body following a postulated core-melt accident. These are the first design conditions to be imposed by NRC for mitigation of the effects of postulated core-melt accidents in over ten years and represent a substantial departure from past licensing practice.

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RISK MANAGEMENT: INTEGRATION OF SOCIAL AND TECHNICAL RISK VARIABLES INTO SAFETY ASSESSMENTS OF LWR'S

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ABSTRACT

A risk management methodology is developed here to formalize the acceptability levels of commercial LWR power plants via the estimation of risk levels acceptable to the public and the integration of such estimates into risk-benefit analysis. Utility theory is used for developing preference models based on value trade-offs among multiple objectives and uncertainties about the impact of alternatives. The method involves reducing the various variables affecting safety acceptability decisions to a single function that provides a metric for acceptability levels. The function accomondates for technical criteria related to design and licensing decisions, as well as public reactions to certain choices.

Discussion of the method focuses on the dynamic nature of the assessment process, on specific requirements and on problems associated with (1) familiarization with the terminology and motivation for the assessment, (2) verification of independence assumptions concerning preferences, (3) assessment of the trade-offs among attributes, (4) assessment of the individual attribute utility functions, and (5) checking for consistency and modifications.

INTRODUCTION

Policy planners, safety analyists, and other decision makers involved in the area of thermal reactor safety and licensing have increasingly been concerned with public perceptions of risks associated with nuclear power and with long delays in licensing process due to intervention of special interest groups. The degree of public acceptance of safety, site and environmental suitability and other factors in the licensing process, ranging from strong support to firm opposition, can significantly affect the successful implementation of any nuclear energy development plan. Similarly, an understanding of factors affecting the public acceptance can lead potentially to greater citizen involvement in public decision making and thus make licensing decisions politically more palatable.

In order to expedite and improve the licensing process, the use of standard plant designs and previously designated sites has been proposed. Nevertheless, this is unlikely to reduce the delays caused by potential intervenors unless they have participated in the process prior to the utility's notification to NRC of their intent to file application for licenses to construct and operate a nuclear power plant. In fact, earlier participation of special interest groups could accelerate the licensing process and eliminate any possible cancellation of, or regulatory or judicial actions against plant construction or operation. However, public involvement in decisions related to technical aspects of designs, site designation or environmental data gathering is not usually practical. In such cases, however, risk criteria may be developed by either direct public participation or by assessment of public attitude taking into consideration the psychological factors shaping public perceptions of risk. These criteria could provide measures to assist in the evaluation of technical issues or could be combined with the technical aspects to provide a decision acceptable to the various concerned parties.

The risk management methodology developed here is based upon employing a formal decision model which uses as input estimates of risk levels acceptable to the public (e.g., soft values) together with the integration of such estimates with technical considerations (e.g., hard facts). (In reality, this division is more illusory than real since technical risk estimates are often "best-guess" extrapolations). The most suitable decision model available currently which is capable of accomodating for public perception measures is the multiattribute utility (MAU) theory.

The MAU approach provides a viable means for integrating technical attributes, cost parameters and results of public attitude assessments. Recent development in the application of the MAU approach via interactive communication [1] has provided a facility for active participation of representatives of special interest groups in the decision making process. The interactive MAU model is a flexible, low-cost analytical package that allows public agencies and private groups to conduct their own analyses of major options or to participate with the utility company and/or public officials (e.g., NRC) in a reconciliation decision process.

MULTI-ATTRIBUTE UTILITY MODEL

Multi-Attribute Utility (MAU) theory is adapted here for developing preference models based on value trade-offs among multiple objectives and the uncertainties about what the impact of any alternative will be. The MAU method has been applied to a host of decision problems including nuclear power plant siting [2] and public preferences in selection of energy alternatives [3].

Although a decision analysis of a specific issue must always be adapted to the problem at hand, it will generally involve the following key elements: (1) structuring the decision problem, usually in the form of a "decision tree", showing branch points with branches for alternative modes of action (decision nodes) or for sets of possible outcomes or consequences (change nodes); (2)

estimating the likelihood of uncertain outcomes; (3) characterizing the possible outcomes in terms of parameters or attributes which measure the achievement of desired objectives; (4) valuing outcomes in terms of appropriate criteria and establishing preferences to guide tradeoffs among conflicting objectives; and (5) combining alternatives, likelihoods, attribute values and preferences to determine the most preferred course of action.

The Multi-Attribute Utility method explores the decision maker's preferences, value trade-offs, attitudes toward risk, and his feelings about uncertain consequences. These all go into a utility function which is used to rank order the various alternative actions available to the decision maker.

Some advantages are:

- Operates under conditions of uncertainty
- Exploits decision maker's own feelings
- Provides means for reconciliation of differing views
- Does not require converting all attributes to some common scale of measurement (such as money) and accomodates for diverse units of measurement
- Capable of integrating technical criteria and public reactions to certain choices
- Possible to apply to group decisions
- Has been successfully applied to very complex decision problems
- Very useful for attributes which are difficult to quantify

The following description indicates by an example how one may put the theory into practice in the licensing decision making process. Specifically, a utility function may be assessed over a circumscribed set of attributes or factors which have been identified with public perceptions of risk and benefits associated with nuclear power. These may include psychological risk factors, economic and technological benefits, socio-political risks, and environmental and physicial risk factors. In addition, a number of technical attributes which are of concern to designers and licensing authorities, such as emission control, general plant requirements, material specifications, site characteristics, and characteristics of structures, components, and systems may be used. For each attribute a measure is established to determine the range of possible impacts of any of the alternatives. The range covers the worst to best levels of each attribute and the units of measurement are either subjective (e.g., for psychological risk factors) or objective (e.g., for radioactive emission).

The purpose of the decision process is to obtain an objective function which indicates the relative ranking of the importance of various technical and social components as well as risk attitudes toward components in the overall acceptability criterion. The optimal decision would be one that maximizes the overall acceptability function. However, since the decision problem involves uncertainties, the process involves trade-offs among the different attributes. Thus, the multi-dimensional nature of the problem is simplified by assessing specific quantities or qualities of the individual attributes and then synthesizing components into an overall acceptability, or utility, function.

Multi-Attribute Utility theory has, for instance, been used to help decision makers determine which R & D program should be funded to improve LWR licensability since there are several competing programs and limited funds. There are many facets to each program which must be considered, each program having its own particular strengths and weaknesses. With the MAU approach, characteristics are divided and subdivided until the programs are more easily described by using these more easily quantified measures. Broad and vague aspects are thus not permitted on a large scale. Single-dimensional utility theory is used on these elemental characteristics and these utilities are combined using either an additive or multiplicative function to arrive at an overall program utility. In the additive form, the overall utility function, U, is expressed in terms of the individual utilities, u_i, of attributes, x_i, i = 1, 2, $\cdots$ n as

$$U(x_1, x_2, \cdots x_n) = \sum_{i=1}^{n} k_i u_i(x_i) \qquad (1)$$

where $\sum_{i=1}^{n} k_i = 1$

and k_i is a weight representing trade-offs among the attributes. Equation (1) follows much the same form as the "attitude formation model" [4] which has been developed by Fishbein and Ajzen and which makes a clear distinction between beliefs, attitudes, and behavior as variables with different determinants. The attitude Y(x) towards an object x may be defined as

$$Y(x) = \sum_{i=1}^{n} b_i z_i$$

where i refers to a specific attribute, b_i is the strength of belief which links the attitude object to attribute i, z_i is the value of attribute i, and n is the number of salient beliefs which are currently within the span of attention. The belief component represents knowledge or opinions about the attitude object and the evaluative component is a measure of affect or feeling. An independent measure of the attitude in question may be reliably obtained using the semantic differential technique [5]. The responses of individuals can be aggregated to examine the response of a given social group.

The attitude formation model has been applied [6] to assess public attitudes toward nuclear power, using factor analyses. One of the values of the factor analytic approach is that it simplifies the examination of a large number of belief items used in questionnaires by grouping the most highly intercorrelated belief items into a small number of basic factors.

In the multiplicative form, the utility function is

$$1 + KU\ (x_1, x_2, \cdots, x_n) = \prod_{i=1}^{n} (1 + Kk_i u_i\ \{x_i\}) \qquad (2)$$

where K is a solution of

$$1 + K = \prod_{i=1}^{n} (1 + Kk_i)$$

At the lowest level, $u_i(\cdot)$, represents a utility function for sub-attribute x_i (often linear, exponential, logarithmic, etc.) All of the sub-attributes for a particular attribute are combined using either Eq. (1) or (2) above to get an attribute utility value. Then, all the attribute values for a particular measure category are combined using either Eq. (1) or (2) above to get a measure category utility value. Finally, all of the measure category utility values are combined to get an overall utility value. Each program is evaluated in this manner, and the resulting values are ranked to give program preferences.

Preliminary comparisons of program evaluations illustrate the effects of individual user input on overall acceptance, or utility, estimates. Table 1 indicates that most of the discrepancy in ranks among programs evaluated stems from the differences in scaling weights given to the individual measure categories. Specifically, and not surprisingly, the measure category labelled institutional issues, and including group perception factors, was heavily weighted by the psychologist member of the team of decision makers, while given less importance by the engineer member of the team.

The individual programs can be, and have been similarly decomposed at various levels (measure categories, attributes, sub-attributes) to compare and contrast the inputs of various decision makers in the group.

MAU analysis thus has demonstrated applicability to issues surrounding the licensing process. Not only can the system accomodate both subjective (psychological) and objective (technical) risk factors, but it also addresses attitudes toward risk in general. Thus, for example, analysis can reflect the decision maker's risk aversion (i.e., the desirability of an alternative with uncertain outcome is less than the desirability of its expected values) or risk proneness (i.e., the reverse).

Table 1. Comparison of Utility Evaluations of Measure Categories Using Two Different Weights; For Selection of R & D Programs to Improve LWR Licensability

Program						
Measure Category	Evaluator*	A	B	C	D	Weights (k_i)
1	E	.4715	.3895	.3716	.3404	.40
	P	.3867	.3614	.3028	.3084	.50
2	E	.7435	.2943	.8158	.6683	.30
	P	.7810	.2151	.8083	.6745	.25
3^+	E	.4846	.4132	.4370	.4679	.20
	P	.5522	.5208	.6020	.7401	.45
4	E	.5724	.4309	.6686	.5257	.10
	P	.4333	.2944	.4765	.3980	.15
Overall	E	.5658	.3698	.5476	.4828	
	P	.6003	.4623	.5970	.6154	

$^+$Institutional issues

*E: Engineer, P: Psychologist

INTERACTIVE MAU

A computer package has been developed to simplify the many computational difficulties involved in the above approach. The program consists of several modules to assist the decision maker to do a complete utility analysis to rank options. The main input for the user is in constructing utility functions at the lowest level and estimating certainty equivalents and in constructing scaling constants. Most modules create files for use by other modules as the analysis progress. An important module, of course, is that which inputs values describing each option.

The computer package "MAUP" provides a collection of interactive computer programs in modular form, designed to maximize ease of use. Previous packages to use decision analysis and multi-attribute utility theory interface easily with only those thoroughly acquainted with the theory, but MAUP is written to be user-oriented so that it can be employed by individuals without detailed knowledge of the underlying mathematical complexities of the theory. The user is only required to respond to simple questions concerning preferences for various alternatives. The user's responses to these questions (under certain behavioral assumptions) imply the mathematical form of a personal utility function, which is single-valued criterion for the evaluation and ranking of alternatives. The program can accommodate for results of psychometric analysis of public responses to questionnaires or surveys. MAUP has been written in FORTRAN IV and tested on IBM 360. Modifications are anticipated to allow it to run on DCD 7600. Several features of the program require the use of an interaction terminal with CRT display.

CONCLUSION

If an acceptable risk problem, such as a licensing decision, receives a thorough decision analysis, the alternative with the greatest expected utility is the alternative whose risk is acceptable to the decision maker(s). Many applications of MAU will involve group decisions and will be particularly useful for accomodating public input in the decision process, either by specification of public perception variables or by actual inclusion of citizen representatives in the decision making process.

Research has shown [7] that intuitive rankings of two opposing groups initially showed strong disagreement over alternative solutions to a coastal zoning problem. When rankings were generated by a simplified form of decision analysis, the disagreements generally disappeared. While the use of formal decision analysis might also lead to polarization and increased conflict among participants in the decision process, it is suggested that, if used early in the licensing process, the method can contribute greatly to the identification and clarification of values, beliefs, and risk factors that may alleviate significantly later stages of potential conflict.

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THE PREDICTION OF ACCIDENT SEQUENCE PROBABILITIES IN A NUCLEAR POWER PLANT DUE TO EARTHQUAKE EVENTS

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ABSTRACT

This paper presents a methodology to predict accident probabilities in nuclear power plants subject to earthquakes. The resulting computer program accesses response data to compute component failure probabilities using fragility functions.

Using logical failure definitions for systems, and the calculated component failure probabilities, initiating event and safety system failure probabilities are synthesized. The incorporation of accident sequence expressions allows the calculation of terminal event probabilities. Accident sequences, with their occurrence probabilities, are finally coupled to a specific release category.

A unique aspect of the methodology is an analytical procedure for calculating top event probabilities based on the correlated failure of primary events. In order to underscore this aspect, an example is provided where the top event probability is first calculated assuming independence between primary events and then computing this same probability for different values of correlation between the primary events.

INTRODUCTION

This computational procedure has resulted in the design of a computer program called SEISIM, which is presently at the programming stage of development. An overview of the procedure is presented graphically in Figure 1. SEISIM's modules operate sequentially, accepting input which characterizes the dynamic responses of power plant structures and internals to an earthquake.

In general, the code will access structural dynamic response data to compute failures using expressions called fragility functions (univariate or multivariate cumulative probability functions of component or structural capacity as a function of local stress or other response characteristic). These computations result in a collection of component and structural failure probabilities. Next, logical failure definitions which, coupled with the prior computed structural and component failure probabilities, enable the synthesis of initiating event and safety system failure probabilities. Finally, by incorporating the logical descriptions of accident sequences, terminal event probabilities are calculated as a result of the initial set of earthquake responses.

A basic feature of the procedure is the distinction and treatment of random and modeling uncertainty. Random uncertainty, as implemented in the program design, relates to variabilities in basic parameters which are beyond the control of the analyst. In contrast to random uncertainty, modeling uncertainty is contributed by uncertainty in the distributions or models which could be reduced by better modeling or more complete data. Modeling uncertainties can be heavily correlated with each other; for example, biases in design procedures that influence several components.

COMPUTATION OF FAILURE

A feature of the methodology is the computation of failure probability of each component with consideration of the statistical correlations between component strengths and the correlations between computed responses. This is accomplished by using the multivariate normal (or lognormal) distribution.

Let the peak measured response at the point of interest in the structure or component be designated by R, let the capacity of the structure or component be designated by F.[1] Then failure occurs when either (Figure 2)

$$R>F \text{ or } R/F>1 \tag{1}$$

The choice of $R>F$ or $R/F>1$ depends upon the assumption for the probability distribution of R and F. If both R and F are normally distributed, $R>F$ is appropriate. If both are lognormally distributed, then $\ell nR>\ell nF$ ($R/F>1$) is appropriate. The method, at this time, is constrained to these two options. The lognormal distribution is more appropriate to use because of its properties ($0<R<\infty$, $0<F<\infty$) and will be used for the remainder of this development.

For a single variate, let $Z = R/F$. Then if $R>F$, $Z>1$ and $0<\ln Z<\infty$. Assuming independence between ℓnF and ℓnR, $\mu_{\ell nZ} = \mu_{\ell nR} - \mu_{\ell nF}$ and $\sigma^2_{\ell nZ} = \sigma^2_{\ell nR} + \sigma^2_{\ell nF}$.

For the multivariate case, covariance matrices must be developed for ℓnF and ℓnR where $\{\ell nF\}$ and $\{\ell nR\}$ are vectors of values representing corresponding peak responses and capacities at various points within the system. The covariance matrices are arrays containing all of the variances and covariances of the vectors [1]. Using the lognormal representation for responses we have

$$\left[\Sigma_{\ell nR}\right] = \begin{bmatrix} \sigma^2_{\ell nr_1} & \sigma_{\ell nr_1\ \ell nr_2} & \sigma_{\ell nr_1\ \ell nr_3} & \cdot \\ & \sigma^2_{\ell nr_2} & \sigma_{\ell nr_2\ \ell nr_3} & \cdot \\ \text{Symmetric} & & \sigma^2_{\ell nr_3} & \cdot \\ & & & \cdot \end{bmatrix} \tag{2}$$

where $\sigma^2_{\ell nr_i} = \int_{-\infty}^{\infty} \left(\ell nr_i - \mu_{\ell nr_i}\right)^2 f(\ell nr_i)\, d(\ell nr_i)$

$$\mu_{\ell nr_i} = \int_{-\infty}^{\infty} \ell nr_i\, f(\ell nr_i)\, d(\ell nr_i)$$

$$\sigma_{\ell nr_i\ \ell nr_j} = \int_{-\infty}^{\infty}\int_{-\infty}^{\infty} \left(\ell nr_i - \mu_{\ell nr_i}\right)\left(\ell nr_j - \mu_{\ell nr_j}\right) f(\ell nr_i, \ell nr_j)\, d(\ell nr_i)\, d(\ell nr_j)$$

$f(\ell nr_i)$ and $f(\ell nr_i, \ell nr_j)$ are univariate and bivariate normal distributions of the logarithms of R_i and R_j.

The values in (2) must be developed by a joint statistical analysis of the peak responses at each location. It can be expected that the covariance will generally be positive and demonstrate a high correlation. This is due to the commonality of the earthquake forcing function acting on the entire structure. The reason that 100% correlation will not exist is due to the variability of structural responses as a result of variations in the earthquake spectra and in structural response properties.

A similar covariance matrix exists for the capacity of the component or structure at the response points. Covariance elements (off-diagonal terms) in this matrix frequently will be zero except in the cases where e.g., they represent correlation between identical manufactured components.

Following the development for a single variate, the vector ℓnZ can be developed, such that

$$\left\{\mu_{\ell n\ Z}\right\} = \left\{\mu_{\ell n\ R}\right\} - \left\{\mu_{\ell n\ F}\right\} \tag{3}$$

and

$$\left[\Sigma_{\ell n\ Z}\right] = \left[\Sigma_{\ell n\ R}\right] + \left[\Sigma_{\ell n\ F}\right] \tag{4}$$

At this point we have the complete description of a multivariate lognormal distribution capable of being used to compute the marginal or joint probability of failure of any one or group of components within the system. Thus this description can be used to compute properly the joint probabilities of failure defined by the minimal cut sets resulting from the fault and event tree definitions of the system. The first step in this procedure is to form marginal distributions represented by the elements of the cut sets. For example, consider the computation of $P[(\ell nZ_i>0)\wedge(\ell nZ_j>0)]$. The covariance matrix for the marginal distribution is

$$\begin{array}{ccccccccc}
 & & \text{column } i & & & \text{column } j & & \\
\left[\begin{array}{c} \\ \\ \\ \\ \\ \\ \\ \end{array}\right. \sigma^2_{\ell n\ z_i} & \cdot & \cdot & \cdot & \cdot & \cdot & \cdot & \left.\begin{array}{c} \\ \\ \\ \\ \\ \\ \\ \end{array}\right] & \\
\cdot & \cdot & \cdot & \cdot & \cdot & \cdot & \cdot & & \\
\cdot & \cdot & \sigma^2_{\ell n\ z_i} & \cdot & \cdot & \sigma_{\ell n\ z_i\ \ell n\ z_j} & \cdot & & \text{row } i \\
\cdot & \cdot & \cdot & \cdot & \cdot & \cdot & \cdot & & \\
\cdot & \cdot & \cdot & \cdot & \cdot & \cdot & \cdot & & \\
\cdot & \cdot & \sigma_{\ell n\ z_i\ \ell n\ z_j} & \cdot & \cdot & \sigma^2_{\ell n\ z_j} & \cdot & & \text{row } j \\
\cdot & \cdot & \cdot & \cdot & \cdot & \cdot & \cdot & &
\end{array}
\Rightarrow
\begin{bmatrix} \sigma^2_{\ell n\ z_i} & \sigma_{\ell n\ z_i\ \ell n\ z_j} \\ \sigma_{\ell n\ z_i\ \ell n\ z_j} & \sigma^2_{\ell n\ z_j} \end{bmatrix} = \left[\Sigma_{ij}\right] \tag{5}$$

The joint probability is obtained from the integration

$$
\begin{aligned}
P\left[(\ell n\ Z_i>0)\Lambda(\ell n\ Z_j>0)\right] &= \int_0^\infty\int_0^\infty f(\ell n\ z_i,\ell n\ z_j)\ d(\ell n\ z_i)\ d(\ell n\ z_j) \\
&= \frac{1}{|\Sigma_{ij}|^{\frac{1}{2}}(2\pi)}\int_0^\infty\int_0^\infty \exp\left[-\frac{1}{2}\left\{\begin{matrix}\ell nz_i-\mu_{\ell nz_i}\\ \ell nz_j-\mu_{\ell nz_j}\end{matrix}\right]^T\left[\Sigma_{ij}\right]^{-1}\left\{\begin{matrix}\ell nz_i-\mu_{\ell nz_i}\\ \ell nz_j-\mu_{\ell nz_j}\end{matrix}\right\}\right]d(\ell nz_i)\ d(\ell nz_j)
\end{aligned}
\qquad (6)
$$

The most significant aspect of the above discussion is that joint, as well as univariate, failure probabilities can be computed. This will correctly handle the problems of correlated failure. A limitation, perhaps, is the assumption of lognormality (or normality) throughout.

GENERAL METHODOLOGY

Reference to Figure 3 outlines the general approach employed by SEISIM. The module names are those identified in the computer code. The following provides a brief outline of the purpose and computations of each module.

Although the PREPROCESSOR module is shown as the first in the sequence of computations, there is a module which takes in all user input data and performs various data checks prior to running the code. These user inputs are shown asterisked in the flow diagram and for convenience are defined at the module where they are first operated on.

The PREPROCESSOR inputs [RQ], which is a matrix of peak responses measured at various points on the reactor structure and at components. For each earthquake, defined by its peak acceleration and corner frequency (measured at the site), some twenty or thirty time histories of site response will be used by prior computer runs of structural dynamics models to provide input for obtaining the statistics of local peak responses. Each time history will be weighted according to its expected occurrence frequency and this vector {WTH}, used to weight [RQ], will be used to calculate the weighted mean vector $\{\mu_R\}$, weighted standard deviations $\{\sigma_R\}$ and weighted correlations of peak response quantities, $[\rho_{R_iR_j}]$.

The PREPROCESSOR also accepts as input, the vector [UFF] of fragility (capacity) data related to the points on the structure, and components, where the responses to the earthquake have been calculated. The mean fragilities are accepted as input ($\{\mu_F\}$ is a subset of [UFF]), but the fragility standard deviations can either be input directly ($\{\sigma_F\}$ a subset of [UFF]) or percentile data can be input enabling $\{\sigma_F\}$ to be estimated.

The next two modules COV and SUMF compute the covariances of responses Σ_R and of fragilities Σ_F. The inputs required for these calculations are provided by the PREPROCESSOR, plus user input $[\rho_{F_n F_m}]$, a matrix of fragility correlation coefficients. The same subroutine is used by COV and SUMF to compute these covariance matrices. SUMF further calculates the mean vector and covariance matrix of $\{Z\}$. The mathematics of these computations, and their necessity for computing failure probabilities of structures and components, was described in the previous section.

The heart of the computer code is embodied in the module PFAIL. This module performs all the failure probability calculations for structural members and components subject to seismic loading. By defining the Boolean logic for initiating events (caused by the earthquake), safety related systems and accident sequences, in terms of groups of minimal cut set expressions, the required failure probabilities can be calculated. These minimal cut set expressions (or correlated primary events) defining the failure modes of systems [GCS], initiating events [ECS] and accident sequences [QCS], are required as inputs.

Each sequence is tagged as to a specific release category {IREL} [2]. A subroutine within PFAIL searches for the appropriate mean vector and covariance matrix of the fragility-related primary events of each cut set term, from $\{\mu_Z\}$ and $[\Sigma_Z]$. 'Random' type failures of components are included in the minimal cut set expressions, e.g., unavailability due to test and maintenance. The matrix of random failure probabilities for components is included in [RFP]. (Within a given cut set, fragility-related failure and random failure are mutually exclusive.) The required multivariate failure probability calculations are performed by a further subroutine within PFAIL [3]. This subroutine operates by successive numerical iterations and has been modified to perform the required manipulations, within the context of this program, as efficiently as possible. The routine presently has the capability to calculate a fragility-related cut set probability of dimension ten.

The large dimension integrals are rather time consuming to compute. However, the computation time for a large size analysis can be reduced dramatically by recognizing that, for different accident sequences, many minimal cut set terms will be repeated. For most cases, beyond the bivariate, it is probably better to store probabilities than to re-calculate them. The cut off point in terms of efficiency between storage and search has still to be investigated.

The final computations performed by PFAIL are the terminal event probabilities $\{P(Q^*_\ell)\}$ where

$$P(Q^*_\ell) = P(Q_\ell \wedge IEQ(\ell) \wedge EQ_n) = P(Q_\ell / IEQ(\ell) \wedge EQ_n) \cdot P(IEQ(\ell)/EQ_n) \cdot P(EQ_n) \qquad (7)$$

$Q_\ell / IEQ(\ell) \wedge EQ_n$ is sequence ℓ given initiating event $IEQ(\ell)$ and earthquake n.

$IEQ(\ell)$ is the initiating event upon which sequence Q_ℓ is conditioned.

$P(EQ_n)$ is the user input probability of earthquake n.

Matrix [X] stores integer tags of the accident sequences sorted into release categories. Matrix [ASP] stores the associated probability estimates $\{P(Q^*_\ell)\}$.

The module ASTAT calculates release category probabilities (storing them in {RCP}) by summing the terminal event probabilities in each release category.

The next module, DSEQ, searches for and stores the dominant sequences, both in terms of probability (within each category of expected release) and according to various weighting schemes (across all categories). The weighting schemes are user defined and could be, for example, the fraction of expected core inventory released in each release category for different isotopes. The weighting option allows a comparison between high probability-low release events and low probability-high release events.

Once the dominant sequences have been determined, DCAG makes use of the results to determine the dominant components and component groups (safety systems). The measure of dominance presently employed in the program is calculated in the following way:

$$\mathrm{SUMCP}(i,\ell') = \sum_{\text{All } k/C_i \varepsilon k} P(\mathrm{QCS}_{\ell' k}/\mathrm{IEQ}(\ell')\wedge \mathrm{EQ}_m) \tag{8}$$

Equation (8) yields the sum of the conditional probabilities of minimal cut sets of dominant sequences ℓ' which contain component C_i. $\mathrm{QCS}_{\ell' k}$ is the kth minimal cut set term in dominant accident sequence ℓ' and $\mathrm{IEQ}(\ell')$ is the seismically induced initiating event on which accident sequence ℓ' is conditioned.

The sum of the actual probabilities of the minimal cut sets as defined by equation (8) is

$$\mathrm{SUMP}(i,\ell') = \mathrm{SUMCP}(i,\ell')\cdot P(\mathrm{IEQ}(\ell')/\mathrm{EQ}_n)\cdot P(\mathrm{EQ}_n) \tag{9}$$

Finally, the measure of component C_i's contribution to the probability of release category n is

$$\mathrm{AP}_{in} = \left\{ \sum_{j=1}^{\mathrm{IDS}(n)} \mathrm{SUMP}(i,\ell') \right\} \div \mathrm{RCP}_n \tag{10}$$

where IDS(n) is the number of dominant sequences ℓ' in release category n and RCP_n is the probability of release category n.

Once the dominant components have been determined, DCAG computes the dominance ranking of primary input variables according to their impact on the dominant components. (Primary input variables are such variables as soil stiffness, soil damping, structural stiffness, structural damping, etc.; values of which have been used in the structural dynamic analysis to compute structural and component responses.) A matrix of partial derivatives of responses at dominant components with respect to primary input variables is calculated. The largest value of these partial derivatives, for a given component, is considered to yield the primary input variable with greatest impact. The method requires a regression of responses on primary input variables. An 'F' test will be performed to measure the significance of any lack of fit and, hence, the significance of the conclusions as to dominant primary input variables.

After DCAG has completed its runs, the user has the option to stop there, or make use of the DERIV module. DERIV will measure the sensitivities of release category probabilities to shifts in the mean values of response and fragility (modeling uncertainties) and to shifts in the standard deviations of response and fragility (random uncertainty). The suggested shifts are those for the dominant components from DCAG.

At present, a shift in the mean value of response or fragility for a component is assumed to have no effect on the respective correlations. For the computation of numerical partial derivatives of release category probabilities, re-runs of the program require only re-calculation of the cut set probabilities (and hence accident sequence probabilites) affected by the relevant shifts.

RESULTS

A sample problem with the following Boolean expression for the top event failure modes

$$f = A \cup (B \wedge D) \cup (B \wedge E) \cup (C \wedge D) \cup (C \wedge E) \qquad (11)$$

was used to run some test cases to illustrate the methodology for calculating terminal event probabilities. The primary events A through E are, for this example, all fragility related.

The input data was as follows:

$$\mu_Z = \left\{\begin{bmatrix} -37.5 \\ -34.0 \\ -10.15 \\ -30.0 \\ -22.1 \end{bmatrix}\right. \begin{array}{l} A \\ B \\ C \\ D \\ E \end{array} \qquad \sigma_Z = \left\{\begin{bmatrix} 15.0 \\ 20.0 \\ 7.0 \\ 25.0 \\ 17.0 \end{bmatrix}\right\} \qquad \rho^{(i)} = \begin{bmatrix} 1 & & \rho^{(i)} \\ & \ddots & \\ \rho^{(i)} & & 1 \end{bmatrix} \qquad \begin{array}{l} \rho^{(1)} = 0 \\ \rho^{(2)} = 0.25 \\ \rho^{(3)} = 0.75 \end{array}$$

An estimate of the upper bound on the terminal event probability was calculated using equation (11). This result was compared with the estimate obtained when all cross product terms were included. The marginal probabilities were calculated to be

P(A)= 0.0062 P(C)= 0.0735 P(E)= 0.0968
P(B)= 0.0446 P(D)= 0.1151

The following were obtained for the terminal event probabilities based on different values of correlation.

ρ →	0	0.25	0.75
UPPER BOUND	0.0312	0.0579	0.1532
ALL TERMS	0.0291	0.0448	0.0658

It is too early in the program to state, with any degree of certainty, what levels of correlation can be expected between structural component responses and between structural and component capacities. It is to be expected however that the covariance matrix of structural and component capacities will be much more sparse than that for structural and component responses.

Even for relatively low levels of correlation (0.25 throughout) the expected terminal event probability is approximately twice that for the case where structures and components are assumed to behave independent of each other.

As the level of correlation increases, it becomes evident that the upper bound, as expressed by equation (11), becomes a less satisfactory approximation to the actual value expected when all cross product terms are included. For higher levels of correlation therefore, and for higher marginal probabilities, a better estimate of the upper bound will be required. Also, as the number of minimal cut sets describing a terminal event sequence increases, the upper bound calculated here will become progressively poorer. The authors have been pursuing an alternative approach to a solution which can be used as these conditions become limiting. This approach will be the subject of a later paper.

ACKNOWLEDGEMENTS

This report was prepared as an account of work on the Seismic Safety Margins Research Program (SSMRP), subcontracted to the J.H. Wiggins Company by Lawrence Livermore Laboratory. The SSMRP is sponsored by the U.S. Nuclear Regulatory Commission under the auspices of the U.S. Department of Energy. The authors wish to express their appreciation to the Nuclear Systems Group at LLL for their participation in the conceptualization phase of SEISIM's development. The authors also wish to express their thanks to Ms. Jean Gasca and Mr. Bruce Kennedy of the J.H. Wiggins Company for their tireless efforts towards obtaining results suitable for incorporation into this paper.

FOOTNOTE

[1] Resistance or capacity is sometimes referred to as fragility. The presentation of fragility is usually a cumulative probability distribution of failure as a function of load or response. In this paper, the probability density function is used to describe variability in capacity, but when integrated, it is equivalent to the fragility curve.

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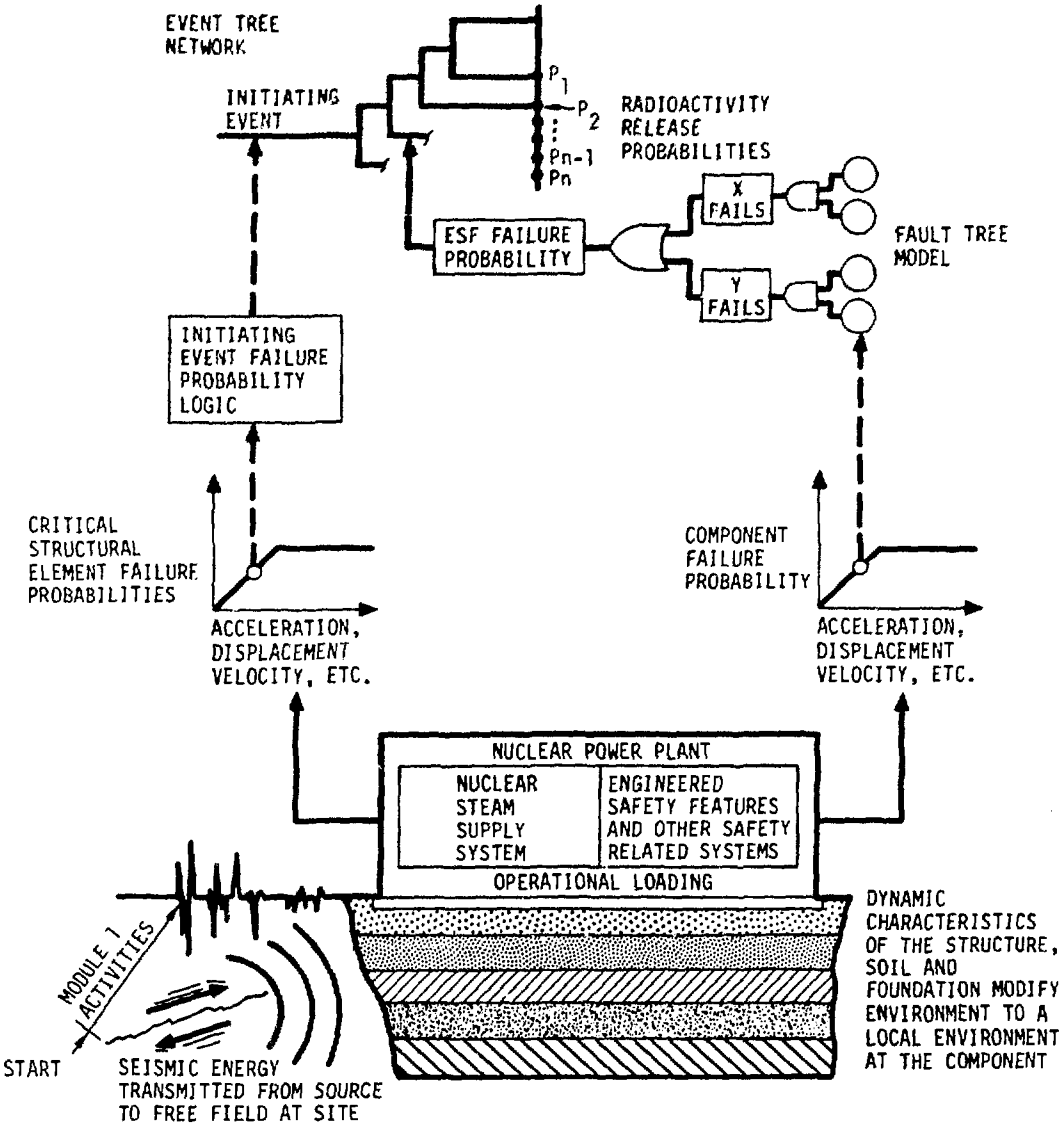

Figure 1. Graphical Description of Computational Procedure

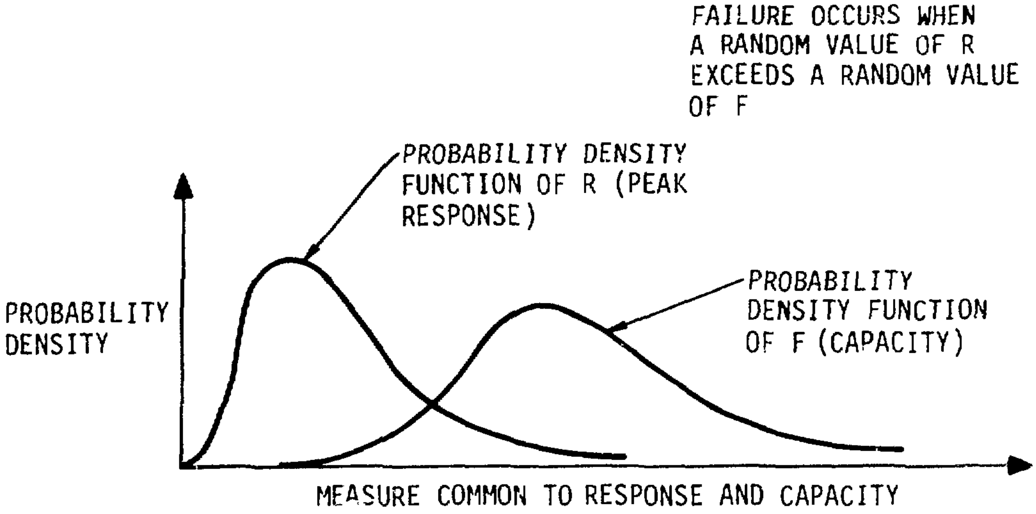

Figure 2. Density Functions for Response and Capacity

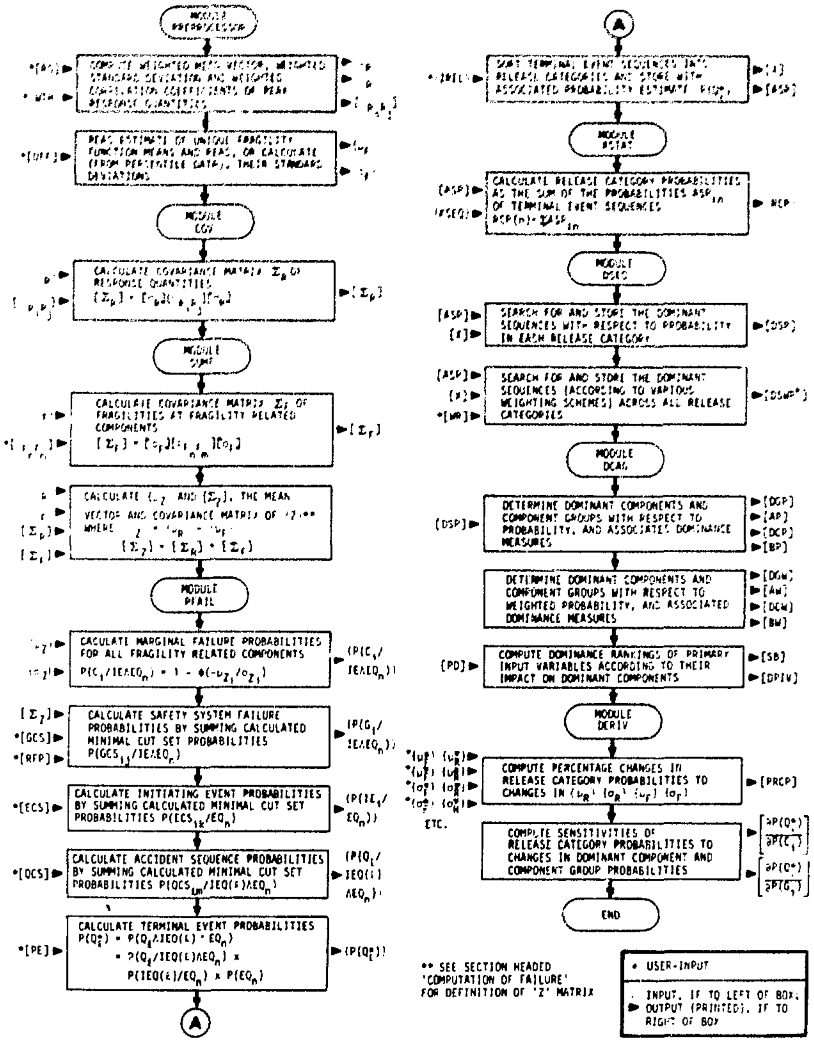

Figure 3. General Computational Approach

A JUSTIFICATION OF THE STATIC COEFFICIENT OF 1.5 FOR EQUIPMENT SEISMIC QUALIFICATION

by

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ABSTRACT

Regulatory Guide 1.100 requires that the use of 1.5 as a static coefficient for equipment analysis be justified. In this paper, the mathematical derivation of the static coefficients for three types of equipment is presented. It is shown that the maximum static coefficient would not exceed 1.5, even for very conservative assumptions.

1.0 INTRODUCTION

Regulatory Guide 1.100 (Ref. 1) on the seismic qualification of Electrical Equipment for Nuclear Power Plants states that:

> "As indicated in Section 5.3, 'Static Coefficient Analyses,' a static coefficient of 1.5 is used for equipment analysis to take into account the effects of both multifrequency excitation and multimode response. The use of 1.5 as a static coefficient should not be considered acceptable unless justified by analysis."

The basis of the above statement is that:

> "There is no adequate evidence presented in Section 5.3 to substantiate the validity of a static coefficient of 1.5, or one greater or less than 1.5 in its application to equipment analysis."

The need to use the static coefficient in a static analysis is not limited to the electrical equipment. For instance, for mechanical equipment such as flexible tanks and heat exchangers, the response spectrum technique is generally used. However, the use of a static analysis is not only feasible but also economically desirable. As for vavles supported by piping systems, it is customary to use a static analysis if and when the valve can be designed to be rigid. Only when it is flexible, a response spectrum analysis may be necessary. This generally requires that a time history analysis be conducted for the piping system so that the applicable response spectrum can be developed at the valve location. A static analysis using a static coefficient multiplied by the valve acceleration (from a piping analysis with a simple valve representation) is more than adequate. This requires, of course, that a static coefficient can be justified. To date, no such justification can be found in the literature.

It is the intent of this paper to provide some theoretical basis to justify the use of a static coefficient for certain types of equipment. Generalization to other types of equipment is also possible.

Based on a paper presented by the author and published in the 5th SMiRT Conference in Berlin (Ref. 2), the mechanical equipment can be classified into three types according to its dynamic characteristics.

The first is the rigid equipment for which the natural frequencies are such that no response amplification is possible from its base input motion. Such equipment is particularly suitable for static analysis. A static coefficient of 1 multiplied by the maximum base motion will suffice.

The second type is the flexible equipment with only one predominate mode. A typical representation of this equipment is vertical tanks and heat exchangers. It has been shown in a previous paper presented and published by the author (Ref. 3) that such an equipment can be simplified by a two mass model. Along any given direction the dynamic response can be obtained from the two modes. Also, the fact that the second mode is predominately rocking motion indicates that essentially only one mode will produce significant dynamic response from the earthquake type of input motion. Using a closed form solution, an equivalent static coefficient of 1 can be proven to be the conservative upper bound value for such an equipment.

The third type of equipment possesses multiply contributing modes. The dynamic repsonse of this equipment can be divided into two parts. The first part is the predominate modal response. The second part is the residual modal contribution. With some simplification, the equivalent static coefficient for this type of equipment can be shown to be no greater than 1.5.

Finally, discussions are also provided for any necessary extension of the present formulation to other types of equipment.

2.0 THEORY

The mechanical equipment can be classified into three types according to its dynamic characteristics. The first is the rigid equipment for which the natural frequencies are high such that no response amplification is possible from its base input motion. The second type is the flexible equipment with only one predominate mode. Finally, the third type is the equipment which has multiply contributing modes. Each type of equipment has its distinctive dynamic characteristics and should be treated separately.

In what follows, derivation of the appropriate static coefficients is made for each type of the equipment.

2.1 Rigid Equipment

This equipment has a sufficiently high fundamental natural frequency which is not excited by the input motion at its base. For a seismic input at free field, such frequency is generally indicated by the (rigid) frequency at which the response spectral acceleration converges to the maximum ground acceleration. For the equipment supported by a concrete structure, the rigid frequency needs to be adjusted to the frequency value at which there is no further response (or amplification) from the building motion. This value is, in general, less than the 33 Hz based on the rigid frequency of the free field motion. For this type of equipment, not only each equipment modal spectral response above this rigid frequency will be the same as the maximum ground motion, the total response of the rigid equipment is the same as the maximum ground acceleration. That is, a rigid equipment essentially moves with its base and has no further amplification. Therefore, a static coefficient of 1 will suffice.

2.2 Flexible Equipment With One Predominate Mode

Vertical tanks, heat exchangers, and filters which are supported either by a skirt or columns at the bottom part of the vessel can be flexible. Experience has shown that this equipment has only one flexible mode along one horizontal axis. For this equipment, only one mode could be excited by the response spectral peak value. The remaining modes will all be subjected to the constant maximum base acceleration. Although any analysis conducted for this equipment should include the modal amplification for the flexible mode, the equipment need not be subjected to very conservative analysis which includes the multiple mode effect and the effect of closely spaced modes.

In fact, in Ref. 3, the author has shown that such an equipment can be treated as a two degrees of freedom system (Fig. 1). The fundamental mode is basically a translational bending mode while the second mode is primarily a rocking mode. In addition to that, the second mode (rocking mode) is generally at substantially higher frequency than the fundamental mode, the rocking mode has essentially a zero participation factor. Therefore, only one mode will contribute to the total response. The response force at each of the mass points 1 and 2 can be written as the following:

$$F_1 = m_1 \frac{m_1 \phi_{11} + m_2 \phi_{12}}{m_1 \phi_{11}^2 + m_2 \phi_{12}^2} \phi_{11} S_1 \tag{1}$$

$$F_2 = m_2 \frac{m_1 \phi_{11} + m_2 \phi_{12}}{m_1 \phi_{11}^2 + m_2 \phi_{12}^2} \phi_{12} S_1 \tag{2}$$

where F, m, ϕ, and S are the force, mass, mode shape coefficient, and the acceleration response value, respectively. The subscripts for F and m and the second subscripts for ϕ indicate the mass number these quantities associate with. The first subscript for ϕ and the subscript for S are the modal numbers.

It is reasonable to assume that

$$m_1 = m_2 = m \tag{3}$$

The total response force of the equipment becomes

$$F_1 + F_2 = 2 m \alpha S_1 = m S_1 \left(1 + \frac{2\phi_{11} \phi_{12}}{\phi_{11}^2 + \phi_{12}^2}\right) \tag{4}$$

where α is the static coefficient.

By normalizing the mode shape coefficient at the second mass point to 1, that is,

$$\phi_{12} = 1 \tag{5}$$

the static coefficient becomes

$$\alpha = \tfrac{1}{2} + \frac{\phi_{11}}{1 + \phi_{11}^2} \tag{6}$$

Taking partial differentiations of α with respect to ϕ_{11} and set it equal to zero, one arrives at

$$\phi_{11} = 1 \tag{7}$$

which results in an upperbound value for α when it is substituted into Eq. (6). That is,

$$\alpha_{max} = 1 \tag{8}$$

This upperbound static coefficient is obtained for shear force. For bending moment, the exact value will be dependent on the height of the equipment, specifically, the height ratio of the tank versus the support (such as the skirt if it is a tank). To establish an upper bound static coefficient value, three cases will be evaluated.

1. H = 0
   This indicates that the tank has a flat bottom and is sitting on a mat.

The total moment at the base is

$$F_1 \frac{3}{4} \ell + F_2 \frac{1}{4} \ell$$

$$= m \ell S_1 \alpha$$

$$= \frac{m \ell S_1}{4} \frac{3 \phi_{11}^2 + 4 \phi_{11} + 1}{1 + \phi_{11}^2} \tag{9}$$

Taking partial derivative of α with respect to ϕ_{11} in Eq. (9) results in the following solution:

$$\phi_{11} = 1.618, \text{ or } \phi_{11} = -0,618 \tag{10}$$

The positive ϕ_{11} yields an upper bound value of α which is

$$\alpha_{max} = 1.059 \tag{11}$$

2. $H = \ell$

   This indicates that the support is of equal height to the tank body. Using the same procedure as in 1 above, one obtains the values for ϕ_{11} as

$$\phi_{11} = 1.180, \text{ or } -0.847 \qquad (12)$$

the larger value is obtained by using the positive ϕ_{11} which results in

$$\alpha_{max} = 1.007 \qquad (13)$$

3. $H = 2\ell$

   This represents an extreme case since the support is not normally designed to have great height.

   Again, using the same procedure as above, one arrives at

$$\phi_{11} = 1.105, \text{ or } \phi_{11} = -0.905 \qquad (14)$$

and

$$\alpha_{max} = 1.002 \qquad (15)$$

From Eqs. (11), (13) and (14) one may conclude that the largest static coefficient for bending moments is when the support is the shortest. However, in view that most of the equipment such as vertical tanks and heat exchangers does have a reasonable height, enough at least to allow the equipment to have a spherical bottom. The equal height assumption used in Case B may be more realistic. Also, the equipment generally has more weight toward the lower part, either by weight of the water or merely by design, that the dynamic motion as reflected by its mode shape will be less pronounced at the top mass. As a result, the static coefficient for bending moment should also be close to (if not below) 1, as in the case for shear force.

C. Flexible Equipment With Multiple Modes

This equipment includes flexible vertical components supported at two or more elevations, horizontal components supported by saddles, and refueling equipment such as cranes.

For this equipment, amplification on each of the flexible modes is possible. Also, more than one mode could be subjected to the peak response motion. It is for this type of equipment that a justification of a realistic static coefficient is difficult to obtain.

It is possible, however, to evaluate some simple cases and attempt to extrapolate from the results to general applications.

Take, for instance, a two-degrees-of-freedom system. The reverse inertia force for each mass can be written as the following:

$$F_1 \leq m_1|k_1|\ |\phi_{11}|\ S_1 + m_1|K_2|\ |\phi_{21}|S_2 \tag{16}$$

$$F_2 \leq m_2|k_1|\ |\phi_{12}|\ S_1 + m_2|k_2|\ |\phi_{22}|\ S_2 \tag{17}$$

where

$$k_1 = \frac{m_1\ \phi_{11} + m_2\ \phi_{12}}{m_1\ \phi_{11}^2 + m_2\ \phi_{12}^2}\text{, and} \tag{18}$$

$$k_2 = \frac{m_1\ \phi_{21} + m_2\ \phi_{22}}{m_1\ \phi_{21}^2 + m_2\ \phi_{22}^2} \tag{19}$$

All nomenclature are the same as defined earlier.
For simplicity, let

$$m_1 = m_2 = m, \tag{20}$$

$$\phi_{12} = \phi_{22} = 1\text{, and} \tag{21}$$

$$S_1 = S_2 \tag{22}$$

The total reverse inertia force for the system becomes

$$\begin{aligned} F_1 + F_2 &= 2\ m\ S\ \alpha \\ &= m\ S\ (1 + 2\ \left|\frac{\phi_{11}}{1+\phi_{11}^2}\right| + \left|\frac{1-\phi_{21}}{1+\phi_{21}^2}\right|) \end{aligned} \tag{23}$$

The bracketed value in Eq. (23) can be shown to be less than 3.2. Therefore,

$$\alpha \leq 1.6 \tag{24}$$

The static coefficient shown in Eq. (24) is extremely conservative. For instance, instead of using absolute sum in Eqs. (16) and (17) square root sum of the squares would be more appropriate. Assuming that the first mode contributes for about 66% of the response, the new α would be

$$\alpha \leq 1.20 \tag{25}$$

More uniform distribution of the modal responses would result in an even smaller static coefficient.

Also, it is highly unlikely that both modes would have the same response spectral (peak) value. Reduction of the static coefficient is possible when the more realistic modal spectral values are used.

The above derivation is based on a two degrees of freedom system. It is possible now, with the results of Eq. (25) established, to extrapolate to systems with residual modes which are high frequency in nature. For conservative purposes, assume that this additional residual modes only affects the response but not the total mass. Assume further that the modal spectral response at these higher modes are the same as at the two fundamental modes. One arrives at the following static coefficient:

$$\alpha < 1.30 \qquad (26)$$

This is only a slight increase from Eq. (25). It is to be noted that the static coefficients in Eqs. (24), (25), and (26) are based on comparison of the total reverse inertia force, but not the bending moments. From the results presented for the one mode predominate system, the static coefficients determined for bending moments are not significantly different from those derived for total shear. Consequently, it is expected that the static coefficients will not differ appreciably either, in the case of the multimode system.

3.0 CONCLUDING REMARKS

A theoretical development has been presented to establish the maximum static coefficients for various types of equipment (Table 1). It has been shown that for a rigid equipment or an equipment with only one predominate mode, a static coefficient of 1 will suffice. However, for an equipment with multiple contributing modes, an upper bound static coefficient of 1.5 may be more appropriate.

4.0 REFERENCES

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TABLE 1

MAXIMUM RECOMMENDED STATIC COEFFICIENTS FOR EQUIPMENT

TYPE OF EQUIPMENT	STATIC COEFFICIENT
Rigid	1.0
Flexible, One Mode Predominate	1.0
Flexible, Multiple Modes	1.5

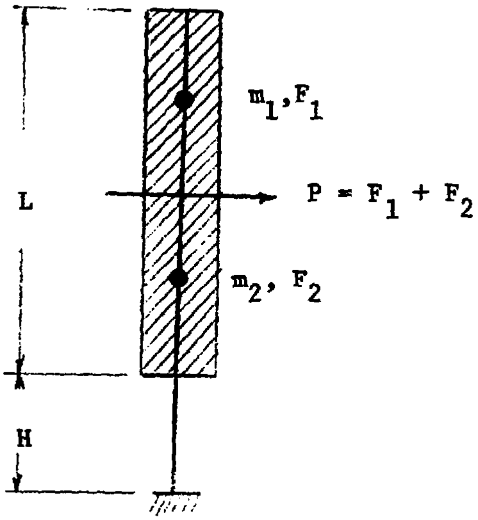

FIGURE 1: SCHEMATIC REPRESENTATION OF A TWO-DEGREE OF FREEDOM EQUIPMENT MODEL.

SEISMIC INSTRUMENTATION OF SWISS NUCLEAR POWER PLANTS

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Summary

Switzerland is an area with moderate seismic activity. Nevertheless, as a contribution towards an improvement in safety, seismic instrumentation of NPP's is justified. This paper gives information, concerning the Plant Seismic Instrumentation Guide which is based on US practice and Swiss conditions. Also described are equipment, functioning, and test results of the trial instrumentation that has been installed.

1. Introduction

1.1 Motivation

Although a catastrophic earthquake is an event of very low probability, it could cause great harm in an area of high population density. Therefore the earthquake risk concerning nuclear power plants is taken into serious consideration in Switzerland. Also, it must be remembered, that an earthquake is a potential cause for common mode failures throughout the plant. Suitable seismic instrumentation is therefore an important factor for determining protective measures to be taken. The requirements are laid down in ref.1.

1.2 Purpose

The requirements of the instrumentation of nuclear power plants are as follows:

- to compare the event data with the design basis data. This information is a help for the decision to shut down the plant in case the event exceeds the design basis values. Furthermore the measurements allow checking the accuracy of the mathematical models and parameters, which were used for designing the plant.
- to register the event characteristics, such as excitation and behaviour of structures, systems and components.
  The response of structures and equipment may be used for recalculation of plant design. Also, the response is used as a data basis for designing new power stations.

1.3 Background

Switzerland is an area with moderate seismic activity. The seismic instrumentation of nuclear power plants is based on the seismic hazard map of Switzerland. The creation of this map was initiated by Swiss Nuclear Safety Authority with support from the Swiss nuclear power plant owners. The consulting engineers Basler & Hofmann have conducted the study. The results were published in 1977 [2,3]

Data from 2800 historically and instrumentally recorded seismic events within the last 2000 years have been evaluated in the study. As an example, Fig.1 shows the accumulated number of events vs. the intensity distribution for

5 time spans. Fig.2 shows the site earthquake intensity with a probability of occurrence of 10^{-4}/yr, which is the design basis for Swiss nuclear power plants.

2. Regulation

2.1 Plant Instrumentation

ASK has issued the guide R-16 "Seismic Plant Instrumentation" on March 12, 1980. This guide defines minimal seismic criteria and instrumentation requirements for operating nuclear power plants. It is based upon US practice [4,5], and upon experience from the seismic instrumentation of the Swiss Nuclear Power Plant Beznau I.
The guide is applicable for plants having seismic ground acceleration up to 0.3 g at the foundation. For plants with higher acceleration, the instrumentation has to be adjusted to accommodate these special conditions.
The requirements for the instrumentation are given in Table I. Deviations from these requirements have to be substantiated and approved by ASK.

Location	Triaxial Time Histories	Triaxial Response Spectra	Triaxial Peak Acceleration	Seismic Switch
1. Free Field	(1)**			
2. Reactor Building				
2.1 Contain. Foundation	1 *	1 *		1 *
2.2 Floor	1			
2.3 Supp. of React. Piping		1		
2.4 Reactor Equipment			1	
2.5 Reactor Piping			1	
3. Adjoining Aux. Building				
3.1 Components Support		1		
3.2 Component or Piping			1	
4. Separate Building				
4.1 Foundation		1		
5. Possible Addit. Instr.				
TOTAL Number of Indications TOTAL Number of Locations: 9	3 (2)	4	3	1

* Control Room Indication
** Not required where negligeable soil structure interaction

Table I: Instrumentation requirements

The characteristics of instruments have to fullfill given specifications. Especially time-history recording equipment must be able to record the complete seismic event, including transients with the associated time identification.
At least the following signals from all instruments mounted on the plant foundation must be displayed in the control room:

- Exceeding of the OBE design acceleration (seismic switch)
- Peak acceleration during the event (time history)
- Exceeding of the design response spectrum for discrete frequencies and damping (response spectrum)

The plant must be shut down, in case of exceeding the ground acceleration or the response spectrum (OBE design values at foundation level of the containment). The plant can only be restarted, after comparison between measured and design values, and with the agreement of the regulatory authorities.
The R-16 also contains minimal requirement for the instrumentation maintenance.

2.2 Regional Instrumentation

At the time of promulgation of R-16 "Seismic Plant Instrumentation" the guide R-22 "Recording of Seismic Strong Motion Data" was under preparation. Support for this activity of providing strong motion instrumentation should be given by at least all utilities with nuclear plants planned or under construction, as well as certain other organizations.
In view of the fact that the Swiss Seismic Service is limited to the measurement of microtremors, it has not been possible to record any strong motion characteristics. Such strong motion measuring capability would allow recording the regional specific relevant engineering data, such as peak acceleration, time history, spectrum of strong earthquakes and attenuation characteristics. These recordings would also help to improve correlations between the engineering parameters and the seismic data such as intensity, depth of earthquake focus, etc.
Owners of nuclear power plants are interested in such design data because NPP's are designed for high seismic spectra. To increase the yield of data, the strong motion instrumentation should be sited in areas of high seismic activity, rather than near the plants.
The Swiss concept consists of a minimum number of approx. 15 recording stations, distributed over the most important seismic areas. The specifications of the instrumentation are as follow: Multiaxial acceleration sensors (2 horizontal, 1 vertical), four channel recorders (3 axis, time), triggering at 0.01 g horizontal and/or vertical acceleration, frequency range 1-30 Hz, and absolute time identification.

3. Instrumentation of Beznau I

3.1 General

The seismic instrumentation of NPP Beznau I was installed in 1977 as a backfitting action according to the draft guide R-16. The seismic instrumentation system of Beznau I comprises the following subsystems: data acquisition, data transfer, seismic signal recording, information display in the control room, and data processing by computer.
The sensors are positioned as follows:
At the foundation of the reactor building, on the maintenance floor of the reactor building, on top of one steam generator, at the fuel pool, and in the auxiliary building. It is planned to install a sensor on top of the reactor building also. These acceleration sensors are connected by rigid supports to the building or to the component.
The electronic equipment is placed in the service building. Threshold - and peak values are displayed in the control room.

3.2 Measuring Equipment and Its Function

The signal flow diagram can be seen in Figure 3:
A seismic event activates the sensor. The generated signal is amplified, converted into PCM-Code and transferred to the central unit. The signal is regenerated and series-parallel converted, and then the signal flows to the

threshold detector. The detector actuates the recorder and the printer under defined conditions via control logic. The signal is also passed through a transient recorder which stores initial portion of the earthquake data, thus avoiding loss of any such initiating information.
A time signal, received on long wave radio band, is also recorded on magnetic tape to allow the comparison of the recorded data with other event recordings.

A high level signal (SSE) or low level signal (OBE) will cause the threshold detector to actuate the absolute time recorder and the alarm system. Concurrently the alarms are displayed on a control room panel.
On the same panel the peak acceleration values can be monitored for each sensor. This information is received from the peak acceleration memory of the central electronic system via the demultiplexer.
It is intended in the future to install instrumentation for on line conversion of the time history into response spectra in order to allow quick comparison with the design basis data.

3.3 Evaluation of Limiting Values

The NPP Beznau I is designed per general practice for concurrent two axial acceleration: vertical acceleration (a_v) and horizontal acceleration (a_h). The sensors however receive accelerations in three axes: Zenith (a_Z), Nord (a_N), East (a_E).
The threshold values that initiate recording therefore must be adjusted as follows

$$a_Z = a_v \qquad \text{and} \qquad \sqrt{a_N^2 + a_E^2} = a_h$$

To avoid the vectorial addition of a_N and a_E the a_h threshold value has been replaced by the following minimum expressions:

$$a_N = \frac{a_h}{\sqrt{2}} \qquad \text{and} \qquad a_E = \frac{a_h}{\sqrt{2}}$$

The threshold values in Beznau I have been adjusted according to the above equations for 3 different levels: initiation of recording, lower alarm level (OBE) and higher alarm level (SSE).
The limiting values of OBE and SSE are different for the different instrument locations and for horizontal and vertical axes. In Beznau I they vary between 0.04 g (the vertical component of the lower limiting value of OBE indicated by the free field instrument) and 0.57 g (the horizontal component of the higher limiting value of SSE indicated by the sensor on top of the steam generator).

3.4 Data Evaluation

The central unit of the seismic instrumentation transfers the data in digital form to the evaluation unit. The evaluation unit then generates an acceleration oscillogram and adds the time identification.
At the present time, further evaluation is performed at a different location. In the future, the data will be transferred on line to the utility's computer center for evaluation.

3.5 In-Service Inspection

The seismic sensors are tested by artificial, automatically controlled excitation which is a new development. The principle is shown in Fig.4. A constant frequency input signal with increasing amplitude magnetizes the magnetic circuit. This causes the seismic mass to oscillate and the output signal is generated by the piezo sensor. Following this automatic test, the results are compared on the evaluation unit with the results obtained originally after initial installation of the test equipment.

4. Tests

Tests have been made to prove the proper function and evaluation capability of the seismic instrumentation [6]. For this purpose, in Beznau I, the acceleration of the upper part of the steam generator B has been recorded by the seismic sensor G3. The acceleration was caused by the starting up of the primary system main cooling water pump B. The conditions for the start of the magnetic tape recorder were modified for this low excitation signal. The recording time was 60 seconds.

The principal results of this testing are given in the attached figures as follows:

Fig.5 shows acceleration diagram for the first 4 seconds of the pump startup. The program SEIPLOT was used for plotting.

Fig.6 shows the power spectrum of the measured vibration within the first 500 milliseconds of the pump startup.

Fig.7 shows the Fourier spectrum of the pump start test. It was calculated with the fast Fourier transformation program FFTREAL and plotted with the program FFTRPLOT.

Fig.8 shows the response spectrum of the test for different damping values. This response spectrum is relevant for engineering and is used by the regulatory authorities as an after-event decision basis for safety measures to be put into effect.

5. Summary Remarks

Because of the safety importance of earthquake induced load conditions in Switzerland, the seismic instrumentation of nuclear power plants is considered to be necessary. Regulations concerning this instrumentation have been recently promulgated. In view of the fact that the trial seismic instrumentation has shown good results, the measures taken so far may be considered worthwhile.

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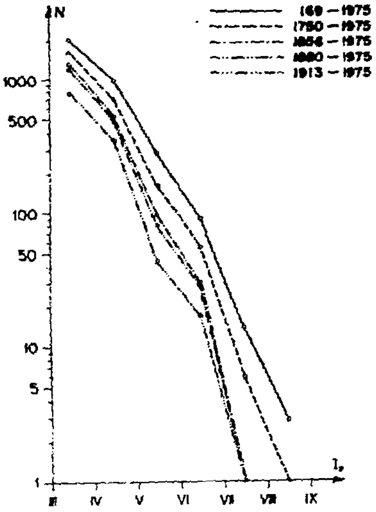


FIG. 1

ACCUMULATED NUMBERS OF EVENTS VERSUS EARTHQUAKE INTENSITY IN SWITZERLAND FOR FIVE TIME SPANS

ID = EPICENTER INTENSITY N = CUMULATIVE NUMBER OF SEISMIC EVENTS

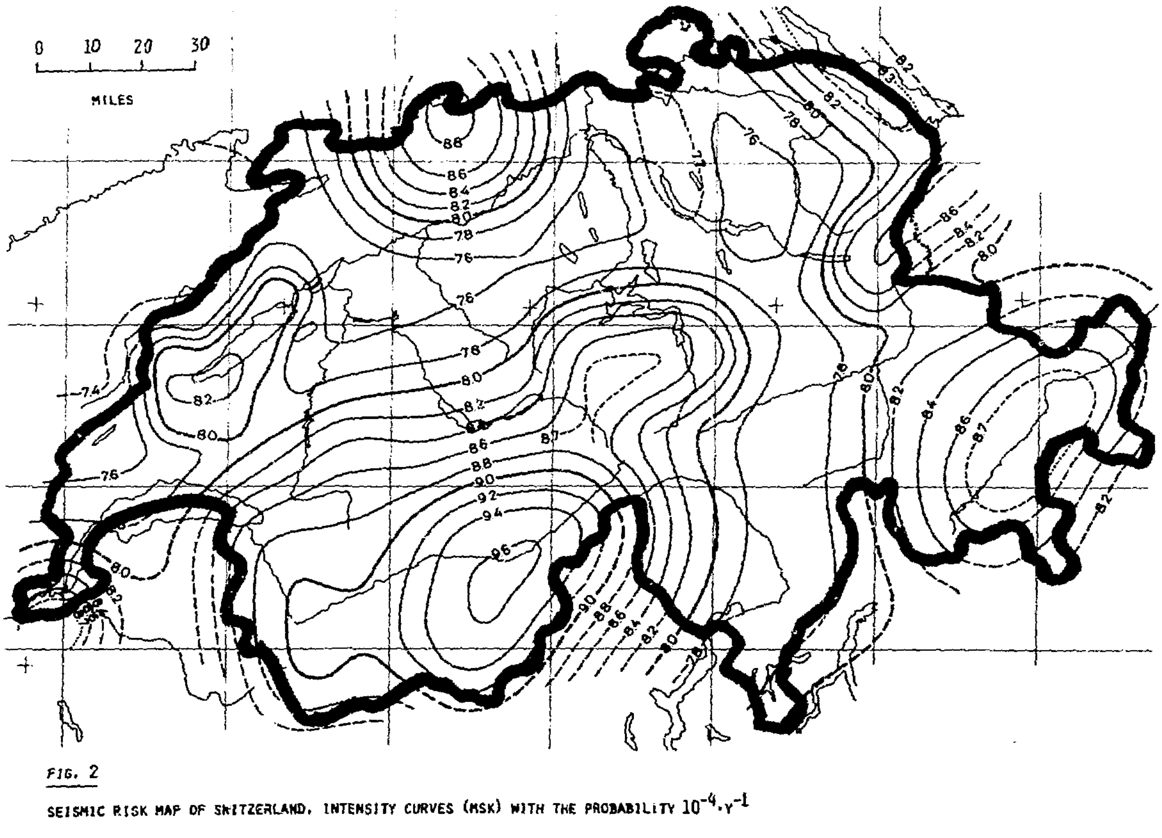


FIG. 2

SEISMIC RISK MAP OF SWITZERLAND, INTENSITY CURVES (MSK) WITH THE PROBABILITY $10^{-4} \cdot y^{-1}$

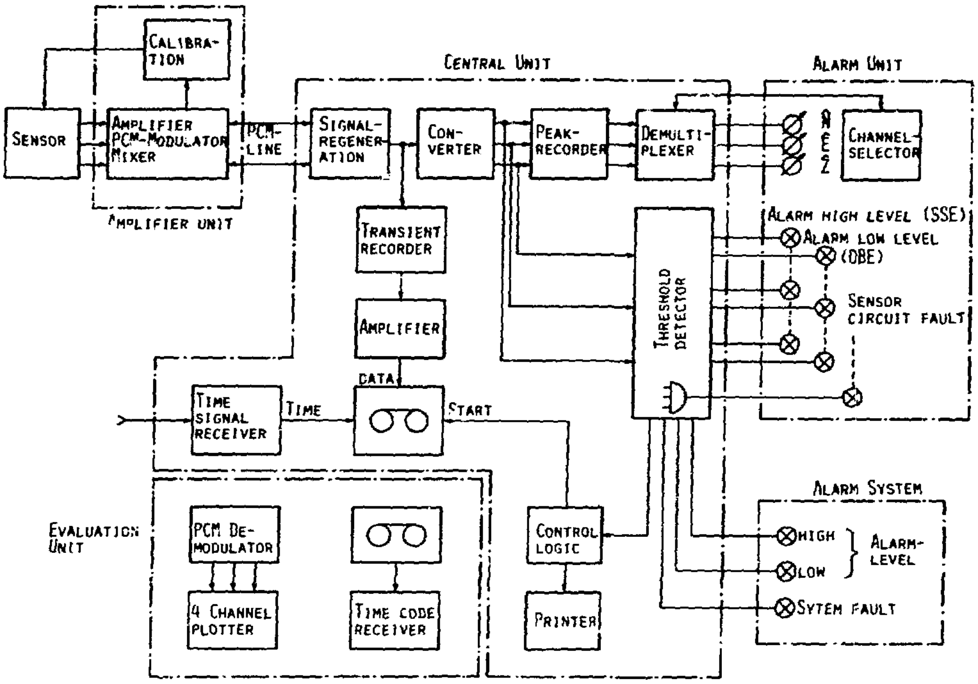
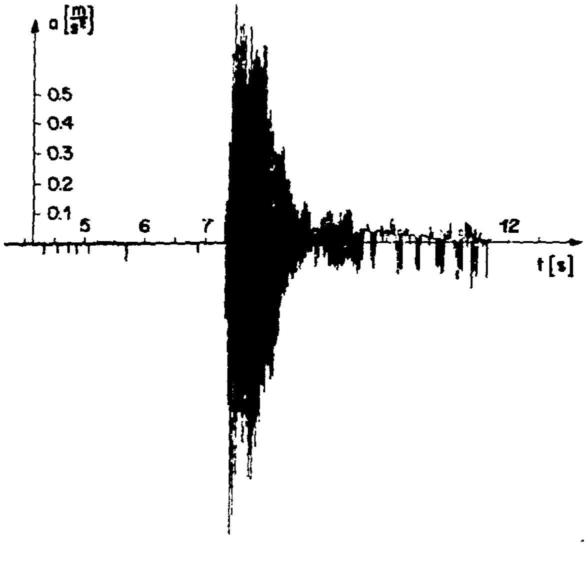


FIG. 3

SEISMIC INSTRUMENTATION BLOCK DIAGRAM



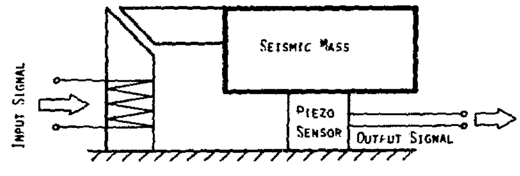


FIG. 4

DIAGRAM SHOWING THE PRINCIPLE OF THE TEST VIBRATION INDUCER

FIG. 5

ACCELERATION DIAGRAM OF THE PUMP STARTUP

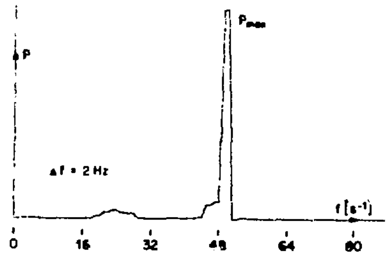


FIG. 6

POWER SPECTRUM OF THE MEASURED VIBRATION WITHIN THE FIRST 500 MS OF THE PUMP STARTUP

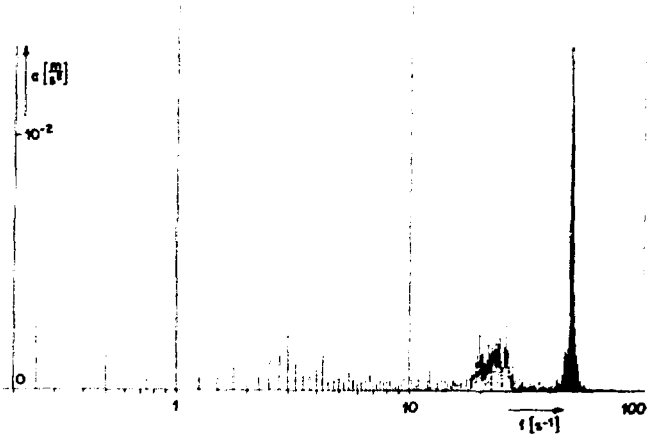


FIG. 7

FOURIER SPECTRUM OF THE PUMP START TEST

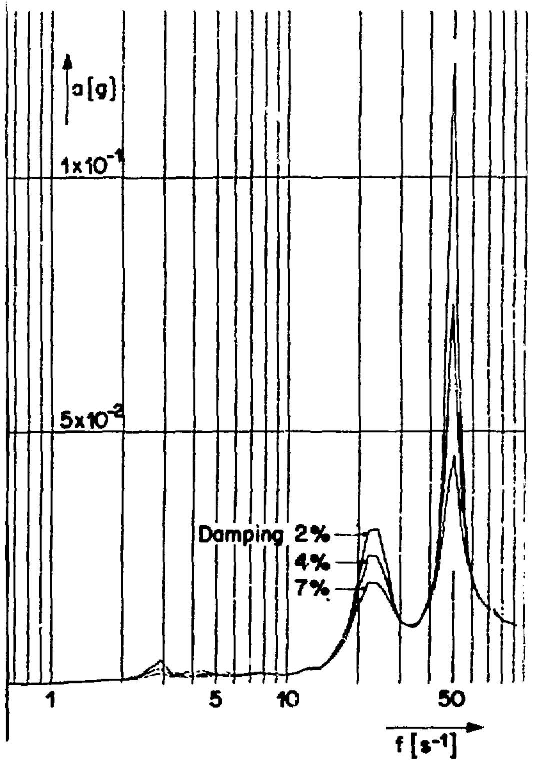


FIG. 8

RESPONSE SPECTRUM OF THE PUMP START TEST

NRC SYSTEMATIC EVALUATION PROGRAM
SEISMIC REVIEW

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ABSTRACT

The NRC Systematic Evaluation Program is currently making an assessment of the seismic design safety of 11 older nuclear power plant facilities. The general review philosophy and review criteria relative to seismic input, structural response, and equipment functionability are presented, including the rationale for the development of these guidelines considering the significant evolution of seismic design criteria since these plants were originally licensed. Technical approaches thought more realistic in light of current knowledge are utilized. Initial findings for plants designed to early seismic design procedures suggest that with minor exceptions, these plants possess adequate seismic design margins when evaluated against the "intent" of current criteria. However, seismic qualification of electrical equipment has been identified as a subject which requires more in-depth evaluation. Plants originally designed to local building codes without explicit seismic design consideration may require substantial retrofitting to achieve acceptable margins.

INTRODUCTION

In October 1977, the Nuclear Regulatory Commission approved initiation of Phase II of the Systematic Evaluation Program (SEP) which consists of a plant-specific reassessment of the safety of 11 older operating nuclear reactors. Many safety criteria have rapidly evolved since the time of initial licensing of these plants. The purpose of the SEP is to develop a current documented basis for the safety of these older facilities by comparing them to current criteria. Phase I of the SEP developed a comprehensive list of 137 topics of safety significance which collectively affect the plant's capability to respond to various Design Basis Events (DBEs). This paper summarizes aspects of the ongoing SEP seismic DBE evaluations and highlights initial findings of the seismic review.

The nuclear power plant facilities under review in the SEP received construction permits between 1956 and 1967. Seismic design procedures evolved significantly during and after this period and through the publication of the Standard Review Plan (SRP) 1975) which represents current analytical and design review criteria along with the Regulations 10 CFR 50, Appendix A, and 10 CFR 100, Appendix A. As a result, the original seismic design bases of the SEP facilities vary in degree from Uniform Building Code considerations (static analysis) up through and approaching current standards (dynamic analysis).

Recognizing this evolution, the NRC has found it necessary to make an assessment of the seismic design safety of the SEP facilities.

GENERAL PHILOSOPHY

The primary objective of the SEP seismic review program is to make an overall seismic safety assessment and where necessary, to improve seismic safety margins, consider backfitting in accordance with 10 CFR 50.109 of the Regulation This Regulation specifies that backfitting will be required only if substantial, additional protection to the public health and safety is required.

To assist in making these decisions, current licensing criteria, as defined by the Standard Review Plan (SRP), are utilized as a baseline to qualitatively assess safety margins. Compliance with all sections of the SRP would certainly imply acceptability; however, this cannot generally be expected in view of the fact that the SEP plants were originally designed to different criteria. Furthermore, demonstrating compliance with specific individual criteria in the SRP is not necessarily an indication of acceptability when considered outside the context of the SRP "package," since individual criteria may not generally control broad safety issues.

The important SEP review concept is to make a determination whether or not the plant meets the "intent" of current criteria considered with respect to the general level of safety these criteria dictate as a "package." For example, current criteria strive to ensure that elements of nuclear power plants remain elastic or nearly elastic in order to assure that they meet their functional requirements. The intent of this goal is important and is maintained in the SEP evaluations. However, seismic resistance does not imply a total absence of permanent deformation. Certain structures, piping, and equipment may suffer damage provided that the entire system can achieve and maintain a safe shutdown condition. Therefore, the SEP evaluations require an assessment of broad safety issues considering the various systems interactions in the context of overall plant safety. Potential accident sequences considering the behavior of the total nuclear power system are evaluated by SEP systems engineers. For example, one major area of review is that associated with possible seismic-induced pipe breaks. The review addresses the subsequent impact on other systems and components, assuming single failures, as well as the possible loss of offsite power with its implications on system safety.

The SEP seismic review process recognizes and attempts to deal with the inherent and often unquantifiable capability of these facilities to resist seismic forces and the conservatisms associated with current evaluation methods. The SEP evaluations attempt to more realistically quantify unclaimed factors which contribute to seismic resistance capability. The current evaluation techniques are sometimes overly conservative because certain energy dissipating mechanisms are not quantitatively considered. Nonlinearities below the threshold of overall elastic response are found to dissipate significant energy and in turn reduce design force levels. Energy absorption below elastic limits is represented through structural damping. Parameters such as damping values have been chosen conservatively because of large uncertainty in their selection. This is justified when the uncertainty is characteristic of the nature of a particular process; however, often uncertainty results from a lack of our knowledge which needs to be refined. Over the years, more structural damping data has become available.[1] Accordingly, this data is considered in the SEP evaluations. The SEP review has utilized structural damping values thought more realistic in light of current knowledge to more accurately reflect the true seismic response.

The SEP seismic review focuses on an assessment of the integrity of the reactor coolant pressure boundary and the capability of essential systems and components required to safely shut down the reactor and maintain it in a safe shutdown condition during and after a seismic event or to mitigate the consequences of such an event. Therefore, emphasis is not given to other components and systems, such as the radwaste systems, which ordinarily fall under the scope of Regulatory Guide 1.29, "Seismic Design Classification." This approach was taken to concentrate efforts on those systems which are most important to safety and minimize the impact on review resources. In most cases, the SEP evaluations should be sufficient to infer the capability of other systems originally classed for seismic consideration such as the radwaste systems because these systems are similarly designed.

The Safe Shutdown Earthquake (SSE) is the only earthquake level evaluated because it represents the most severe safety significant design condition to which the plant may be expected to respond. Present licensing criteria sometimes result in the Operating Basis Earthquake (OBE) (usually 1/2 SSE) controlling the design of various structures, systems, and components because of specified load combinations and lower damping and allowable stress levels. Relief is usually granted when these criteria do not imply real safety issues but rather have operational implications. Since a facility designed to shut down safely following an SSE will obviously be safe for a lesser earthquake, it was deemed unnecessary to investigate further the effects of the OBE. Furthermore, NRC staff requirements for an inspection to evaluate any damage to the plant will follow the occurrence of any major earthquake.

OVERVIEW OF REVIEW PROCEDURES

The 11 SEP plants have been categorized into 2 groups based upon the degree seismic design was originally considered and the quantity of available seimsic design documentation. Two different procedures are in use to review the plants in each respective group. The five later plants, Oyster Creek, Dresden 2, Ginna, Millstone 1, and Palisades, are categorized under Group 1 while the six earlier plants, Dresden 1, Yankee Rowe, Big Rock Point, LaCrosse, Haddam Neck, and San Onofre 1, are categorized under Group 2.

The general review procedures employed are summarized as follows:

Group 1 - Detailed NRC staff/consultant review of existing seismic design documents with supplemental review team studies for spot checking of licensee information and confirmation of review team judgments.

Group 2 - Detailed review of new licensee seismic design evaluations with limited supplemental review team studies for spot checking.

These procedures were developed to optimize the allocation of combined NRC staff and licensee resources and to complete the seismic review within the SEP time frame.

The SEP seismic review was first initiated by conducting a detailed review of the respective plant docket files to provide a core of pertinent information to be used in the review. From the Group 1 plant docket searches, it was concluded that the existing documentation including supplemental information available at the offices of the architect-engineer (AE) and nuclear steam supply system (NSSS) designer would for the most part be sufficient to permit the necessary SEP evaluations. However, due to a lack of available information from all sources for the older Group 2 plants, it was determined that new information should be generated. Therefore, each Group 2 plant licensee was requested by the NRC to initiate a seismic evaluation program to document a more current seismic design basis. Similarly, in limited situations where existing information is not sufficient to adequately document the seismic safety of the newer Group 1 plants, these licensees have been requested to generate new supporting documentation.

The NRC staff review of all of the Group 1 facilities should be complete by the end of 1980. Dresden 2 is complete, while Ginna is nearing completion. Oyster Creek, Palisades, and Millstone 1 are in intermediate stages of review.

The Group 1 reviews necessarily require an evaluation of critical structures to assess the seismic input to equipment. However, the seismic stresses for the structures are not evaluated at every location. Spot checks are made at critical locations, but the evaluations generally end at the force level by way of a comparison to the original design force levels. Stress evaluations in addition to the spot checks are made only when the newer force levels are predicted to be higher than the original design values.

A detailed evaluation of the hundreds of individual components within the Group 1 plants is not made. The evaluations rely upon sampling representative components from generic groups of equipment. This sample is augmented subject to walk-through inspections of the facilities to select additional components based upon a higher potential degree of seismic fragility. These procedures rely in large part upon the expertise of the review team. Accordingly, a team of recognized seismic design experts under the direction of Dr. N. M. Newmark has been organized by the NRC to assist in these reviews.

To date the licensees' Group 2 seismic evaluation programs have concentrated upon the development of seismological, structural, and mechanical acceptance criteria and evaluation procedures. The NRC staff is working with these licensees to develop criteria which are consistent with SEP objectives. Working analytical models are in development and detailed evaluations are beginning. In view of the limited existing seismic design data base for the Group 2 plants, it is expected that the scope of review will be more encompassing.

DETERMINATION OF THE SEISMIC HAZARD

The SEP review includes an intensive evaluation of the seismic hazard at each site. Various deterministic and probabilistic techniques are under consideration to produce site-specific ground response spectra. This information will be used to evaluate the adequacy of the seismic input originally specified for design and newer ground response spectra recently proposed by the SEP licensees for use in the SEP evaluations.

The current methodology for determination of design bases for vibratory ground motion as defined by 10 CFR 100, Appendix A, of the Regulations and the SRP utilize the generic Regulatory Guide 1.60 ground response spectra anchored at a zero-period (peak) ground acceleration of the Safe Shutdown Earthquake (SSE). The peak ground acceleration is established based upon required investigations of the local and regional geology, seismic history, and engineering characteristics of the site. These criteria are deterministic and necessarily require bounding interpretations of important parameters. Generally, this procedure is believed to be adequately conservative; however, it has a tendency to produce variable estimates of the seismic hazard from site to site. The penalties associated with being overly conservative are particularly significant when reevaluating older facilities. Therefore, recognizing the inadequacies with the current approach, the NRC has contracted with the Lawrence Livermore Laboratory to develop various alternative seismic hazard methodologies to explore improved approaches.[2]

Probabilistic seismic risk analysis methodologies provide a unique tool for decision making which can be used for:

1. explicitly keeping track of important parameters and the sensitivity of their variance;

2. quantifying predicted risk in a consistent and uniform manner;

3. estimating relative risk from one seismic hazard level or design level to another, and

4. evaluating in a limited way the required level of seismic resistance capability to meet overall risk goals.

Although these methods are controversial particularly in estimating absolute levels of risk as implied in item 4, regulatory experience suggests their usefulness for the quantitative comparison of relative risk as suggested in item 3. If properly used, this information is valuable in considering whether to backfit a facility to increase seismic safety margins.

The following four approaches have been suggested by LLL for evaluation of all sites except San Onofre 1:

1. Direct statistical evaluation of response spectra from appropriate groups of real earthquakes;

2. Scaling of real spectra to peak acceleration values determined for various risk levels;

3. The Newmark-Hall technique of scaling response spectra to peak accelerations and velocities determined for various risk levels; and

4. Uniform risk technique-scaling spectral ordinates as a function of return period.

Figures 1 through 4 provide examples of output from these methods. Figure 5 shows a comparison of the four methods. An important review objective is to assess whether these various methods yield common spectral predictions in the engineering frequency range of interest.

The NRC will incorporate the LLL information, data submitted by the SEP licensees, predictions by alternative methods, and other pertinent data to reach a decision relative to an appropriate seismic input specification.

The San Onofre 1 licensee has proposed a site-specific spectra developed using ground motion modeling techniques from first principles.

BASES FOR REEVALUATION

The specific SEP reevaluation criteria are documented in NUREG/CR-0098.[3] This document addresses:

1. Selection of the earthquake hazard;
2. Design seismic loadings;
3. Soil-structure interaction;
4. Damping and energy absorption;
5. Methods of dynamic analysis;
6. Review analysis and design procedures;
7. Special topics such as underground piping, tanks and vaults, equipment qualification, etc.

Limitations of space make it impossible to summarize all of these issues.

The reevaluation criteria consider the implications of various levels of damage, short of collapse for critical structures and the failure to function for safety-related equipment. Some active elements of nuclear power plants such as pumps, valves, switch gear, motors and motor control centers must remain elastic or nearly elastic to perform their safety function. This objective is maintained within the SEP; however, it is recognized that pure linear elastic analysis, even up to yield stress levels, sometimes overestimate design loadings because certain nonlinearities are not rigorously considered. For active components and other deformation limited items, increased damping levels are considered to more realistically account for energy absorption in the context of overall linear response. The following table summarizes the damping values recommended by NUREG/CR-0098 for the SEP and those specified in Regulatory Guide 1.61.

DAMPING (% OF CRITICAL)

	R.G. 1.61 (SSE)	SEP Recommended (near yield level)
Reinforced concrete	7	7 to 10
Prestressed concrete	5	5 to 7
Welded assemblies	4	5 to 7
Bolted and riveted assemblies	7	10 to 15
Piping	2 or 3	2 to 3
Cable trays	-	10[1]
Equipment	-	7[1]

For structures or passive components such as pressure vessels, tanks, piping, transformers and heat exchangers, for which their safety functions are to maintain leak-tight or structural integrity, energy absorption in the inelastic range is considered in the SEP through use of the "ductility factor." The ductility factor is the ratio of the maximum useful (or design) displacement of a structure to the "effective" elastic limit displacement.[3] For vital items that must remain functional and nearly elastic, ductility levels of 1.0 to 1.3 are recommended in NUREG/CR-0098. Ductility levels of 3 to 8 can be justified to account for inelastic energy absorption when deformation is tolerable from a safety point of view.

It is noted that care must be taken to ensure that local ductilities are maintained at acceptable levels when addressing ductility at a system level.

PRELIMINARY CONCLUSIONS

Structural - Initial review of the later Group 1 plants suggests that these facilities possess adequate structural margins. On a plant-specific basis, some of the earlier Group 2 facilities may require some retrofitting to attain acceptable structural margins.

Mechanical and Electrical - Equipment functional qualification has been identified as any area that requires additional documentation. In certain cases for the Group 1 plants and in more cases for the Group 2 plants, equipment modifications will be necessary. The issue of anchorage and support of equipment and, in particular, electrical equipment has been identified as an area that requires upgrading. The SEP licensees are addressing this issue through individual inspection and evaluation programs. It would appear that piping and mechanical components are adequately supported for the Group 1 plants; however, the Group 2 plants may require substantial upgrading.

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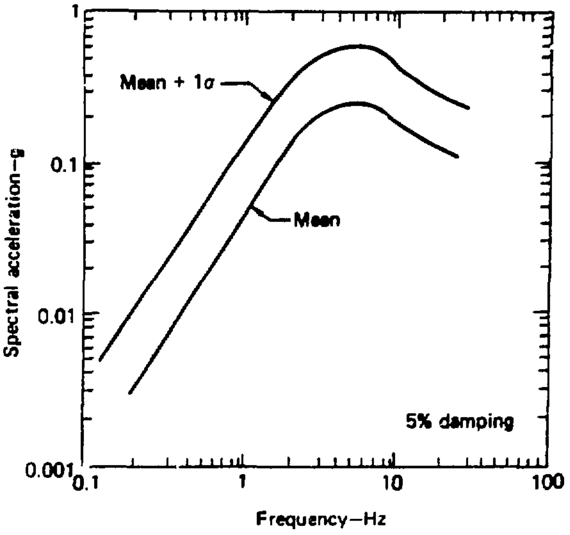

FIG. 1 Real spectra

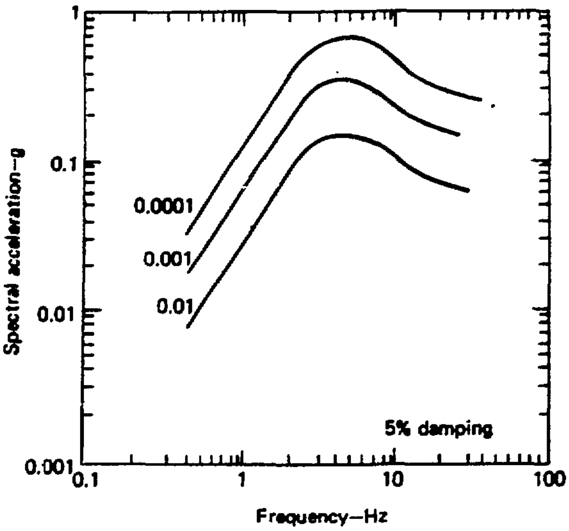

FIG. 2 Real spectra scaled to peak accelerations

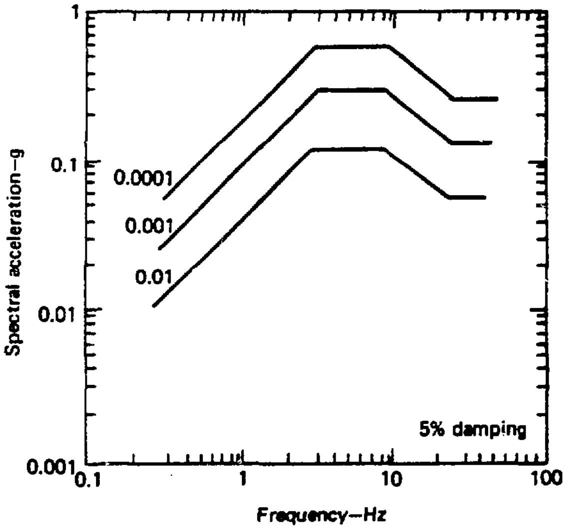

FIG. 3 Newmark-Hall spectra scaled to peak accelerations

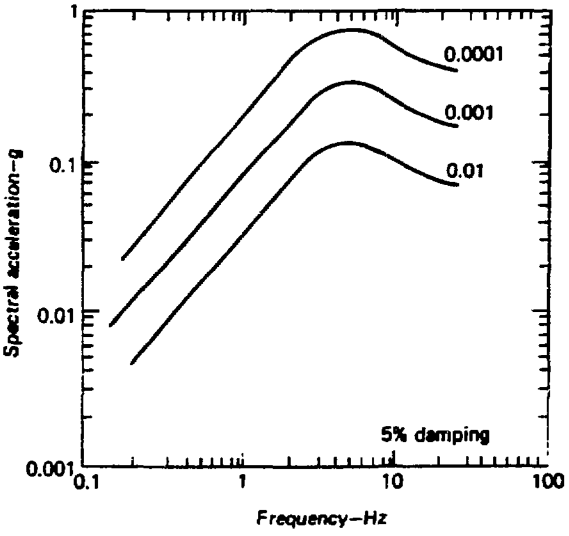

FIG. 4 Uniform hazard spectra

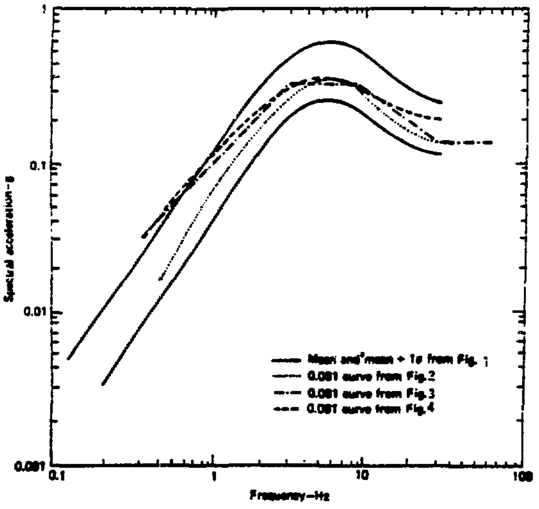

FIG. 5 Comparison of spectra from FIGS. 1-4

A PROGRAM FOR ADDRESSING THE FRACTURE TOUGHNESS REQUIREMENTS OF APPENDIX G to 10 CFR 50

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ABSTRACT

Tests of irradiated weld specimens indicate that some reactor vessel welds may not maintain a C_vUSE of 50 ft-lb throughout the vessel's design life. Because of this, a program was designed to ensure continued safe plant operation through the establishment of realistic fracture toughness criteria and the use of state-of-the-art fracture toughness technology.

Both short- and long-term programs have been developed. The purpose of each program is to satisfy the fracture toughness requirements of Appendix G to 10 CFR 50. The results of the short-term solution show that all B&W 177-fual assembly plants satisfy the fracture toughness requriements of Appendix G for at least 10 calendar years of operation.

INTRODUCTION

This paper describes B&W programs that address the requirements of Appendix G to 10 CFR 50 for reactor vessel weld toughness. A somewhat arbitrary limit of 50 ft-lb has been established in Appendix G for the minimum Charpy V-notch upper shelf energy (C_VUSE) for the entire 40-year vessel design life. If the C_VUSE drops below 50 ft-lb, Appendix G requires that certain actions be taken which can have a significant impact on plant availability.

Tests of irradiated weld specimens indicate that some reactor vessel welds may not maintain a 50-ft-lb C_VUSE throughout the vessel's design life. Hence, a program was designed to ensure continued safe plant operation by establishing realistic fracture toughness criteria and using state-of-the-art fracture toughness technology.

Both short- and long-term programs were developed. The objective of the short-term program is to demonstrate that the requirements of Section V.B of 10 CFR 50, Appendix G (hereafter referred to as Appendix G) are satisfied during the service period that terminates at the tenth calendar year following placement of the nuclear power unit in commercial service. This demonstration is important for two reasons: (1) the 100% volumetric examination required in Section V.C.1 of Appendix G would coincide with the inspection required by Section XI of the ASME Code, and (2) the schedule would allow adequate time to obtain the fracture toughness properties required in Appendix G, Section V.C.2.

The objective of the long-term program is to demonstrate that the beltline region materials have adequate toughness for the continuation of plant operation throughout its design service life. To attain this objective, it is necessary to obtain additional fracture toughness properties of irradiated weld metals and to perform a detailed fracture mechanics analysis on a plant-by-plant basis. The data obtained from the inservice inspections, which will be conducted as required by Section XI of the ASME Code, will be used as input to the fracture mechanics analyses. In addition, one of the long-term tasks is to address the requirements for developing and performing in-place annealing should this be needed as a backup for continued plant operation.

SHORT-TERM PROGRAM

Best-Estimate Design Curves

If the curves of Regulatory Guide 1.99 for predicting the drop in C_VUSE are used, the C_VUSE of the beltline region welds are predicted to drop below 50 ft-lb early in the commercial operating life of the reactor vessel. The prediction curves must be updated to give more realistic prediction values at low fluence levels ($<8 \times 10^{18}$ n/cm^2). The prediction curves should be based on best estimates rather than maximum expected changes. The prediction curves in current use are based on the maximum measured ΔRT_{NDT} or the percent of drop in C_VUSE of all the data available at the time the curves were developed. At that time few data points were available at neutron fluences less than 8×10^{18} n/cm^2. The prediction curves were extrapolated from the range of fluence values for which the majority of data existed (1 to 3×10^{19} n/cm^2) to the lower fluences based on irradiation damage behavior theories.

Recently, data points have become available for fluence values of less than 8×10^{18} n/cm^2. B&W collected these data and has developed an updated design curve for predicting the drop in C_VUSE. The data points and the updated prediction curves demonstrate that the earlier curves are very conservative in predicting the behavior of B&W weld metals.

To develop new design curves, two basic activities must occur. The first is to collect the available weld metal data on welds considered to be representative of the B&W welds. This required a thorough review of the RVSP of pressurized water reactor units now in operation. The second activity is to perform a statistical analysis of the collected data. The results of this analysis were used to draw and justify the best-estimate prediction curves.

Refinement of Neutron Fluence Calculations

To predict the changes in the material properties of the reactor beltline region, an accurate estimate of the neutron fluence to which the region has been exposed is required. The irradiation-induced effects on the material properties are directly related to the neutron fluence; thus, the lower the fluence, the lower the irradiation-induced adjustments to the material properties. In the design of the plants to be evaluated, conservative methods were used to predict the beltline neutron flux. The amount of conservatism in these calculations was benchmarked by recent data obtained from the reactor surveillance capsule analysis programs and used to refine the neutron fluence calculations. Such data, with adequate documentation of its uncertainty, were used to address the previous conservatisms and justify lower fluence values for the service life of each plant.

Characterization of Chemical Composition and Variation of Key Elements

It has been established that the chemical composition of reactor vessel materials strongly influences their response to fast neutron irradiation. The predominant elements in this respect are copper and phosphorus. For the 177-fuel assembly plants, welds were made with Mn-Mo-Ni electrodes coated with a thin electro-deposited layer of copper in the weldments. Earlier work has shown that small variations in this layer probably result in significant variations of this key element within the weldments. Other limited work by B&W indicates that weld joint configuration (e.g., single V, double U) may have an influence on the phosphorus gradient within the weldment. It should be noted that the importance of variations in other elements will become apparent as additional fracture toughness data on irradiated materials become available. A thorough investigation of the chemical composition of weldments is an essential part of the short-term program. Realistic chemical compositions are necessary to predict the irradiation behavior of those weld metals from which material properties will not be available.

Compliance With Reports of Appendix G, Section V.E

Section V.E to Appendix G requires that at least three years before the date when the predicted fracture toughness levels will no longer satisfy the requirements of Section V.B, the proposed programs for satisfying the requirements of Sections V.C (fracture mechanics analysis) and V.D (thermal annealing treatment) must be reported to the Nuclear Regulatory Commission. If the results of the short-term program, as described herein, indicate that the requirements of Appendix G will not be satisfied, the required dates will be established as a part of the short-term program.

LONG-TERM PROGRAM

Irradiation Program

The requirement for additional fracture toughness data on irradiated weld metals is of primary importance. Section V.C.2 of Appendix G requires the additional data. Current programs will obtain fracture toughness data on a number of welds which were selected because of their copper and phosphorus content and unirradiated C_vUSE and RT_{NDT}. These welds will be irradiated in six capsules in power reactors. These data will be supplemented by the data being generated by the regular reactor vessel surveillance programs and by several test reactor programs that include irradiation embrittlement studies.

Fracture Mechanics Analysis

The required procedures to perform fracture mechanics analyses for typical 177-fuel assembly vessels in order to produce the required technical specifications for the power plants will be developed. The objective of this effort is to develop fracture mechanics analytical and test procedures employing current state-of-the-art technology.

Since it is necessary for the procedures to take advantage of the state of the art at the time they are prepared, no decision can be made as to the detailed approach to the problem at this time. Because of this, two alternative approaches will be investigated. The decision as to which approach will be used will be made at the latest possible date in order to reflect the latest advances in fracture mechanics technology.

The technology exists today to apply linear elastic fracture mechanics techniques, such as those outlined in ASME Section XI, Appendix A, in addressing the fracture toughness requirements of Appendix G. However, this approach may place unrealistic requirements on plant operation.

The analytical techniques that would result in the most economically acceptable technical specifications from a plant operation standpoint will be those that account for the slow, stable crack growth that occurs in a structure before the onset of rapid fracture. The technology needed to perform such an analysis (taking into account slow crack growth) is not currently available. Several investigators are currently evaluating and developing this technology, and there is a strong probability that a usable criterion will be developed by the time it is needed. However, the probability that any one program will produce a simple procedure in a timely manner is low. Based on the work performed to date, it appears likely that the procedure that will be developed will be complex, requiring extremely complicated three-dimensional, elastic-plastic, finite element analyses of both the test specimen and the structure.

In-Place Annealing Program

If fracture mechanics procedures should fail to provide a demonstration of adequate toughness margins for continued plant operation, Appendix G requires that an in-place annealing of the reactor vessel be performed. In-place annealing would be performed at an (as yet) unspecified increment in temperature above the operating temperature to recover the fast neutron irradiation-induced damage to the beltline region materials.

At this time, limited work has been accomplished on the response of fast-neutron-irradiated reactor vessel materials to thermal annealing treatments. B&W is closely monitoring industry-wide test programs that are designed to develop the needed data, methods, and requirements for in-place annealing. B&W will then address these results as they apply to B&W operating plants. Hopefully, this will not be required, and the previously described fracture mechanics approach will result in the demonstration of continued licenseability.

RESULTS

Activities of the short-term program have been completed. The predictive techniques and data developed in the program were used to justify continued plant operation to the first required inservice inspection at ten calendar years of plant operation. Specific program results are as follows:

Best-Estimate Design Curves

Using rigorous definitions of C_VUSE, chemistry, and neutron fluence, a broad data base was assembled. A statistical analysis of these data yielded predictive design models (equations) for the percent drop in C_VUSE. The prediction models include as parameters the alloy chemistry and neutron fluence.

Refinements in Neutron Fluence Calculations

Using dosimeter measurements from five reactor vessel surveillance program capsules as benchmarks, refinements in the neutron fluence calculation technique and model were accomplished. The resultant calculations were within 10 percent of the actual measured capsule fluence. The new predictions of neutron fluence through the reactor vessel were reduced by 40 percent over previous generic predictions.

Characterization of Chemical Composition and Variation

Detailed chemical analyses were performed on over 300 samples obtained from archive material weldments. A complete chemistry was then developed for each reactor vessel weld metal by a statistical analysis of the data from the chemical analyses. These chemistries are free of test condition biases and are more extensive in number of elements reported than were the previous chemistries.

To further understand the chemical characterization of as-deposited weld metal, an electron microprobe analysis of a reactor vessel weld was also performed. This has led to a more complete understanding of chemical variations through the thickness of a weld.

Irradiation Programs

All material test specimens (Charpy V-notch, tensile, and compact fracture) for irradiation studies have been fabricated and encapsulated, and the irradiation phase is in progress. Specimen irradiation in the test reactor programs is complete, and partial test data are available. The Charpy V-notch data show less sensitivity to irradiation damage than heretofore expected.

The activities of the long-term program are in progress, and no results are available to present at this time.

J-Integral Elastic Plastic Fracture Mechanics Evaluation of the Stability of Cracks in RPV

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ABSTRACT

The analysis of the stability of cracks in RPV using J-integral fracture mechanics concepts is discussed. An outline of a proposed methodology based on comparing the loading parameters J_{APP} and dJAPP/da with the material resistance parameters J_{MAT} and dJ_{MAT}/da is presented. Two methods to compute J_{APP} and dJ_{APP}/da are analyzed. One of them, based on the line-spring model proposed by Rice, was developed for analyses using elastic-perfectly plastic material behavior. Results for two families of cracks in 228.6 mm thick flat plates are presented for a steel with a 490 MPa yield strength, for a range of crack depths between 68.6 and 182.9 mm and applied stress between 0.2 and 0.9 σ_y. Results obtained using both methods show remarkable agreement. Some J_{APP} values for cylindrical shells of radius equal to 2.286 m and thickness t = 228.6 mm are also shown and compared with flat plate results using the same methods, and with shell and flat plate results using simpler models.

INTRODUCTION

Present ASME Boiler and Pressure Vessel Code fracture methodology is conservatively based on linear-elastic fracture mechanics (LEFM) and, therefore, is limited to the analysis of cracks in vessels at temperatures in, or below, the transition region of the toughness versus temperature relationship. The LEFM approach could be extended for the analyses of cracks in reactor pressure vessels at higher temperatures, in the so called "upper shelf" of the toughness vs. temperature representation, by means of elastic-plastic fracture mechanics based on the J-integral approach [1]. Although alternative approaches could be used [2],[3] none is more general than the JEPFM, or has been as extensively analyzed for possible limitation and applications. An additional advantage is that ASME - LEFM methodology is but a special case of the JEPFM. Furthermore two recent meetings on the subject [4],[5] have concluded that, despite its limitation, the JEPFM is the most promising approach.

JEPFM analyzes the stability of cracks using two loading parameters, J_{APP} and dJ_{APP}/da, and two material resistance parameters, J_{MAT} and dJ_{MAT}/da. The equilibrium and instability conditions are given respectively by:

$$J_{APP} = J_{MAT} \tag{1}$$

$$dJ_{APP}/da > dJ_{MAT}/da \qquad (2)$$

It is sometimes convenient to use the non-dimensional parameters $T_{APP} = (dJ_{APP}/da)\ E/\sigma_y^2$ and $T_{MAT} = (dJ_{MAT}/da)\ E/\sigma_y^2$ = tearing modulus [6]. The material resistance parameters are obtained from a J_R curve, where J_{MAT} is plotted vs. Δa, the crack length increase. They will not be discussed any further since they, and the experimental techniques to obtain them, have been already extensively discussed in the literature [7],[8],[9],[10],[11]. We shall concentrate our analysis, therefore, on new approaches to compute the loading parameters J_{APP} and T_{APP}.

ANALYTICAL DETERMINATION OF LOADING PARAMETERS

Previous elastic-plastic analyses of surface cracks have relied on simple idealizations of crack configurations and material behavior from which instability conditions are estimated, see for example [12],[13]. Correlations between model and experimental results can be secured by judicious selection of some of the model parameters. These approaches may be useful for the analysis and interpretation of specific experimental results, but may lack the generality necessary for extrapolation beyond the experimental data base. In our approach we have used two models originally proposed in the early 70's. One of them was formulated by Erdogan, Irwin and Ratwani (E.I.R.) [14],[15] who analyzed surface and embedded cracks in vessel walls using 8th order shallow shell theory, perfectly plastic behavior in the uncracked ligament and Dugdale strips at the crack ends. A similar analysis was presented later by Krenk [16] who used 10th order shell theory, accounting therefore for shear displacements, and slightly different assumed boundary conditions along the cracks.

The second method is based on the line-spring model of Rice and Levy [17],[18],[19]. The essential idea of the line-spring is that the presence of a part-through surface crack in a thin plate or shell introduces an increased compliance of the body. Physically, this increased compliance manifests itself as an additional "cracked" extension δ and rotation θ of the shell/plate middle surface. Although this additional deformation is typically accommodated over a distance normal to the crack plane of a few plate thicknesses, in the line-spring model this additional deformation is lumped onto the line discontinuity of the plate/shell surface.

It is required that the local values of $\delta(x)$ and $\theta(x)$, where x is a spatial coordinate along the discontinuity be suitably related to the local axial force, N, and bending moment, M, per unit distance along the crack. In the linear elastic regime, these relations can be expressed as

$$\begin{bmatrix} M(x) \\ N(x) \end{bmatrix} = \begin{bmatrix} E_{11}(x) & E_{12}(x) \\ E_{21}(x) & E_{22}(x) \end{bmatrix} = \begin{bmatrix} \theta(x) \\ \delta(x) \end{bmatrix} \qquad (3)$$

where the stiffness matrix E depends on plate thickness, elastic constants, and the relative crack depth at location x, [17],[18],[19].

Although to date, line-spring calculations have been performed mainly for a surface crack in a large elastic plate, the agreement between its predictions for K_I and those of recent 3D finite element analyses is generally quite good, especially considering the one or two orders of magnitude of computing costs by which they differ. Fig. 1 shows a comparision of Raju and

Newman's results using finite element calculations [20] with calculations performed by Parks [21] using the line-spring model. The agreement in K_I is between 2 to 5%, essentially all along the crack front for crack depths to thickness ratios between 0.2 and 0.8, and a/c = 0.2.

In our analyses [22] we have incorporated Rice's [18] suggestions for developing an elastic-plastic line-spring, based upon a nonhardening material model. Following Rice [18] at each coordinate x along the crack projection, a slice is made perpendicular to the plate middle surface. In cross-section the crack is as the single edge crack, with width [illegible] and crack length a. A yield surface $\phi(N,M,a,t) = 0$ for the generalized stresses is constructed from slip line analysis of the single edge crack geometry, and the incremental form of eq. (3) is used

$$\begin{bmatrix} \dot{M} \\ \dot{N} \end{bmatrix} = \begin{bmatrix} E^{el-pl} \end{bmatrix} \begin{bmatrix} \dot{\delta} \\ \dot{\theta} \end{bmatrix} \tag{4}$$

$E_{ij}^{el-pl} = E_{ij}$ for elastic response and

$$E_{ij}^{el-pl} = E_{ij} - \phi_{,m} E_{mi} E_{jk} \phi_{,k} / (\phi_{,p} E_{pq} \phi_{,q}) \tag{5}$$

for plastic loading. The components of the normal to the yield surface are $\phi_{,1} = \partial\phi/\partial M$ and $\phi_{,2} = \partial\phi/\partial N$.

Calculations were performed for a part-through crack in an (otherwise) elastic plate subjected to a farfield pure membrane stress. For this configuration the problem can be reduced to the solution of a pair of coupled single integral equations along the crack.

These equations govern a specific load increment and the entire loading history is accomplished by solving a series of load increments, updating the generalized stress and deformation resultant after each load increment.

It should be noted that the elastic-plastic line spring model could be incorporated into a finite element shell/plate program for analysis of rather general surface crack configurations. For this investigation, however, attention was focused on the large elastic plate because of the simplicity with which it could be investigated using singular integral equations, and the time and manpower limitations of our project.

RESULTS AND DISCUSSION

From the work of E.I.R. crack opening displacements (C.O.D.) = δ_T were calculated for two families of cracks in a flat plate 228.6 mm thick. One family of cracks had a constant surface length 2c = 1371 mm. The other consisted of a series of homologous cracks, with 2c = 6a, as are postulated in the ASME Code procedures. From the C.O.D.'s at the centers of the crack front values of J_{APP} were computed using the formula $J_{APP} = m\sigma_0\delta_T$, where $\sigma_0 = \sigma_y$ and m was assumed equal to one, (see ref. 22) which we believe is consistent with the model of E.I.R. The results of our computations are shown in Figs. 2 and 3 where J_{APP} is given as function of the membrane stress σ. Fig. 2 shows that cracks as deep as a = 182.9 mm by 2c = 1371 mm result in J_{APP} smaller than the $J_{IC} = K_{IC}^2/E = 0.282$ MPa.m for $K_{IC} = 242$ MPa.m$^{\frac{1}{2}}$ at the operating stress of a typical PWR: 157.5 MPa. From Fig. 3 we see that for a homologous family of cracks, at operating stresses, the crack depth required to cause $J_{APP} = 0.282$ MPa.m is practically the same as in the previous example. However, at higher stresses to cause $J_{APP} = J_{IC}$, cracks from the homologous family can be deeper than for the case where $2c = 1371$ mm (Fig. 2).

To analyze the effects of shell curvature, calculations were made for two cracks 114.3 mm deep ($a/t = 0.5$), but of different lengths 685.5 and 1371 mm respectively, in a shell of radius $r = 2286$ mm (90 in.). Fig. 4 shows the results and compares them with the flat plate calculations shown in Figs. 2 and 3. The influence of shell curvature is quite significant at intermediate stresses and very important at higher ones $\sigma > 0.8\ \sigma_o$. However, despite the increased values of J_{APP}, even at $\sigma = 245$ MPa, a 114 mm (4.5 in.) deep crack would be safe if $2c < 1371$ mm, for $J_{IC} = 0.282$ MPa.m. The effects on other crack depths and shapes are similar.

Fig. 4 also includes results for flat plates and shells obtained using the simpler model of Zahoor et al. [12]. Assuming that the E.I.R. model gives more accurate results the use of the model of Zahoor et al., may not always be as conservative as originally intended.

T_{APP}'s were calculated from a plot of J_{APP} vs. a as the one shown in Fig. 5 for the homologous cracks and for one stress level, $\sigma = 441$ MPa. (63 ksi), for the 1371 mm long cracks. The T_{APP}'s so calculated are shown in Fig. 6 where it is seen that, at least for flat plates, $T_{APP} < 10$ except at the higher loads ($\sigma > 350$ MPa) or larger cracks, $a > 160$ mm ($a/t > 0.7$).

Using the line-spring model, J_{APP}, at the center of the surface crack, vs. σ curves were calculated for crack depths $a/t = 0.5$, 0.6 and 0.7, for $t = 228.6$ mm and $2c = 1371$ mm in a flat plate, see Fig. 7, and for $a/t = 0.6$ and 0.7 for $t = 228.6$ mm and $2c = 914$ mm, also in a flat plate, see Fig. 8. The J_{APP} were computed from:

$$J_{APP} = J_{elastic} + J_{plastic} = \frac{\sigma^2 f^2}{E'} + m\sigma_o\delta_T \qquad (6)$$

where $K_I^2 = \sigma^2 f^2$, $f = f(a,t,c)$, K_I *is the linear elastic line-spring calibration* for stress intensity factor, and $m = 2/\sqrt{3}$, since our numerical results suggest that the generalized stress state along the remaining ligament tends towards the membrane state of mid-ligament loading at the edge crack geometry.

In view of the simplicity of our model, however, we have calculated what we believe to be conservative estimates of J_{APP}. We note for example that the idea of adding an "elastic" J which continues to increase quadratically with increasing load parameter may be appropriate in 2D configurations where load continues to rise after first yield due to strain hardening. In our case, however, the increase in load after first yield is principally due to a transfer of load distribution to the larger ligaments toward the crack ends, as the yielded zone spreads from the center plane. Load does not dramatically increase at the center plastic ligament once it has yielded. The local "K_I" or $J_{elastic}$ actually decreases, since M becomes much more negative while N increases very slightly [22].

The results of our line-spring calculations span the linear elastic analysis of surface flawed plates through the fully plastic strip yield results of E.I.R. as shown in Figs. 7 and 8. Indeed, the agreement with the J_{APP} computed from the C.O.D.'s of their membrane model, in the flat plate limit, and our results is quite good. In fact, our more general model tends, in the fully plastic regime, to the membrane stress state which they assumed. This favorable comparison gives additional confidence in the J values for shells which are inferred from E.I.R.'s work.

Before discussing the engineering implication of our results some additional observations about the line-spring model are pertinent. First, the line-spring itself should be most appropriate for large aspect ratio surface cracks, that is 2c >> a. Experience in the linear elastic range suggests that, for predominantly tensile loading, good results can be obtained for moderate ratios of 2c/a of the order of 5. In the plastic regime, the propriety of using the model for lower aspect ratios is undetermined. We emphasize with Rice [18] that a shortcoming of the model, as used, is the neglect of contained yielding prior to reaching the slip line yield surface, although for the problems investigated here, use of such features as a plasticity-adjusted crack length such as $a + (K_I/\sigma_o)^2/2\pi$ for contained yielding could probably be accommodated.

Another important shortcoming of the model is the neglect of yielding at the points where the crack front intersects the free surface. Again, Dugdale zones as "uncracked" line-springs could be incorporated here as in the E.I.R. model.

Finally the drift towards loss in triaxiality indicated by the (M,N) trajectory discussed in more detail in [22], is an important topic which should be investigated further. It is likely that an isotropic hardening version of the yield surface, presently under investigation, would tend to suppress this drift. Also, it would seem likely that small applied positive bending loads would be very effective in this regard as well.

ENGINEERING APPLICATIONS OF RESULTS AND MODEL

Because of the limited space available it would be impossible to discuss in detail many of the applications of our results and of the model used. Only the main trends in the data and the most obvious applications of the model will be analyzed here. Further information could be found in references [21] and [22].

For the purpose of engineering analysis the data could be plotted as T vs. J as shown in Fig. 9 where the (J_{APP}, T_{APP}) points correspond to the homologous family of cracks in a 228 mm thick flat plate. To illustrate the applications of this type of plot (J_{MAT}, T_{MAT}) points obtained from a J_R curve for a 533 B steel 4TCT specimen, tested at 93°C, [9], were used to define the limits of stable (J_{APP}, T_{APP}) combinations. A possibly significant trend in our data can be observed in Fig. 9: although the data represent rather wide ranges of crack depths and applied stresses, the (J_{APP}, T_{APP}) points (including all the points which are not shown) fall within a narrow band extending from $J_{APP} = 7.8$ MPa.m and $T_{APP} \approx 2$, to $J_{APP} = 1400$ MPa.m and $T_{APP} = 20$. Furthermore this plot can be used to convey information concerning the limits of applicability of the J-integral methodology as determined by size requirements, $b > 25\ J/\sigma_y$ or by the ω (or similar) parameters, $\omega = (b/J)(dJ/da) >> 1$, also shown in Fig. 9. Other important trends are shown in Fig. 10 where the (J_{APP}, T_{APP}) points for the 137 mm deep cracks are plotted for four membrane stress levels: 196, 245, 343 and 441 MPa. At two stresses, 245 and 441 MPa, the effects of rate of length growth at the surface, $\Delta 2c$, are illustrated by points corresponding to $\Delta 2c = 12\Delta a$ and $\Delta 2c = 0$. For the homologous family of cracks $\Delta 2c = 6\Delta a$. The trend towards higher T's due to the increased rate of loss of constraint associated with higher $\Delta 2c/\Delta a$ ratios has important implications in structures like RPV where stress and property gradients tend to favor higher ratios. As in Fig. 9, T_{APP} increases monotonically with J showing a strong dependence on the latter. In this plot (Fig. 10), however, the stability could be analyzed by comparison with J_{MAT} vs. T_{MAT} data obtained by F. Loss et al. [23] from J_R curves for submerged arc welded A533-B specimens having high impurity copper level (0.35%). Two weldments were tested with different unirradiated toughness: weld Code V84 with high Cv upper shelf energy, 145 J, (107 ft-lb), and Code V86 with Cv = 108 J, (80 ft-lb). Some of the specimens were irradiated only, I; some irradiated and annealed, A; some were irradiated-annealed and irradiated, AR; some irradiated-annealed-irradiated and annealed, ARA; and, some irradiated-annealed-irradiated-annealed and irradiated, ARAR. For each specimen, or group of specimens, the (J_{MAT},T_{MAT}) points at lower T's and higher J's correspond to 1.5 mm crack extension. To determine the trend of the T_{MAT} vs. J_{MAT} relation the points at higher T's and lower J's were plotted for J_{IC} and the T_{MAT} at J_{IC}. Since Loss [23] observed that the J_R curves exhibited power law behavior the T_{MAT} vs. J_{MAT} should be linear on the log-log representation of Fig. 10. This dependence was used to plot the trend bands for irradiated specimens shown in Fig. 10. The strong trend downwards is also found, but not plotted, with approximately the same slope, for the A and ARA specimens. The closeness of the $(J_{MAT},T_{MAT})_{IRR}$ points to the (J_{APP},T_{APP}) points at the 245 MPa (35 ksi) stress level should be analyzed in the context of the following important facts: a) consistent normalizing factors should be used in the calculations of T_{MAT} and T_{APP} and in the computations of J_{APP} from C.O.D.'s.; b) our J_{APP} and T_{APP} were calculated using the assumption of perfectly plastic material behavior while if hardening, and its dependence on irradiation and temperature, are considered in the models it is expected that the T_{APP}'s can be significantly higher than we computed, as observed by Szabo et al. [24]; c) more research is needed to establish unequivocally under what conditions experimental J_R curves should be obtained to be used in JEPFM analyses of engineering structures; d) NRL irradiated data [23] was obtained under experimental reactor conditions where a $1.5x10^{19}$ n/cm^2 > 1 Mev fluence was achieved in 630 hours at 288°C as opposed to much longer times for the surveillance specimens; and e) the accuracy of the models used to compute J_{APP} and T_{APP} becomes increasingly important since estimation errors cannot be protected against by using generous safety factors.

With regards to the line-spring model our result show its potential as a practical engineering tool to compute J_{APP} and T_{APP} for wide ranges of crack and shell configurations and material behavior. Extensive parametric analysis, as illustrated by Fig. 10, could be performed at low costs, and expected accuracies of the order of 20%, by using the model, including material hardening, in conjunction with a shallow shell computer code. More accurate calculations could be made by incorporating the model into a 3-D elastic-plastic F.E. code. This approach will be also extremely useful to compute J_{APP} and T_{APP} for thick shell configurations as the ones used in the HSST program, where substantial yielding through the thickness of the vessel occurred before the crack became unstable. If this is achieved, two very important objectives would be fulfilled: the validity of the model would be tested against very important experimental results and a more direct analytical link between the HSST model tests and RPV could be established.

CONCLUSION

1) The line-spring model has been proven to be a very convenient analytical tool for the JEPFM analysis of cracks in structures. It includes, as a special case, in the limit of brittle behavior, presently accepted LEFM methodology, providing results comparable in accuracy with those obtained by means of 3-D F.E. methods, but at much lower costs.

2) The present state of development of the model is such that it could be rather easily incorporated into shallow shell or 3-D elastic-plastic F.E. codes.

3) Our limited results on J_{APP} and T_{APP} in flat plates together with NRL results on J_{MAT} and T_{MAT} would indicate that cracks 137 mm deep by 823 mm long in irradiated weldments made from A533-B steel with high copper (35%) content may be close to instability at stresses of the order of 245 MPa (35 ksi). This proximity would require high accuracy in J_{APP} and T_{APP} calculations specially considering the assumptions made in the calculations and the condition under which the experiments were run.

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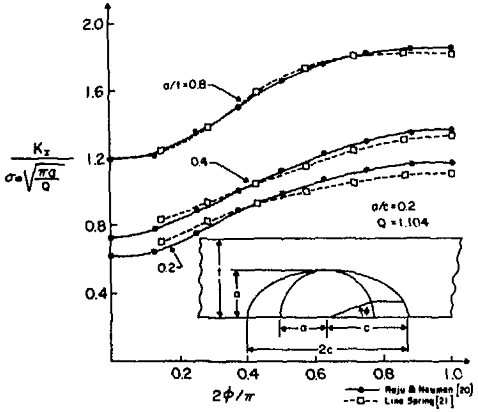

Fig. 1 - Comparison of Normalized Stress Intensity Factor Distribution Obtained from Line-Spring Model [21] and 3-D Finite Element Solutions [20].

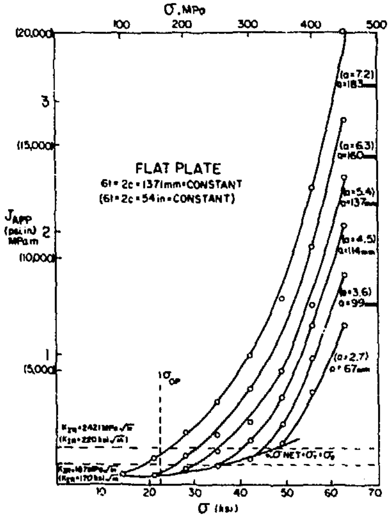

Fig. 2 - Dependence of Japp on Membrane Stress σ for Various Crack Depths a of Constant Length 2c = 1371 mm, in a Flat Plate of Thickness t = 228 mm and σ_y = 690 MPa. [illegible] Model [illegible]

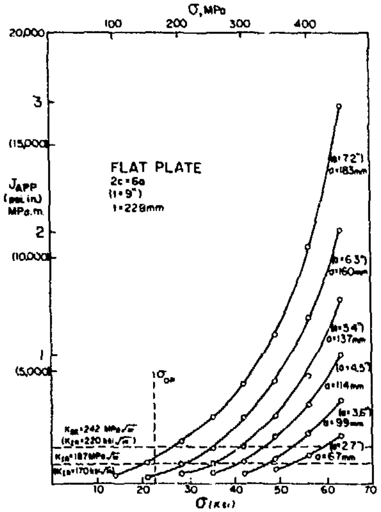

Fig. 3 - Dependence of Japp on Membrane Stress σ for Various Crack Depths a and Length 2c = 6a in a Flat Plate of Thickness t = 228 mm and σ_y = 690 MPa [illegible] Model [illegible]

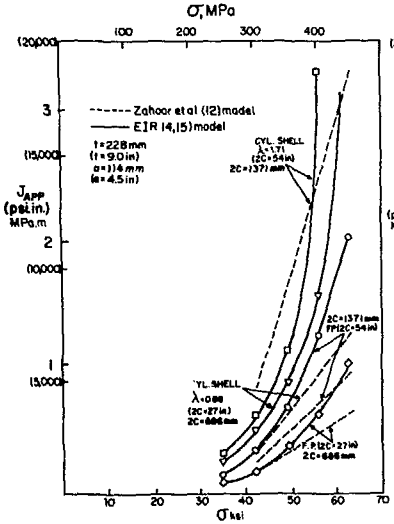

Fig. 4 - Dependence of Japp on Membrane Stress σ for a Crack Depth a = 114 mm (a/t = 0.5) and Crack Length 2c = 1371 mm and 686 mm in a Flat Plate of Thickness t = 228 mm and a Cylinder of Radius R = 3301 mm and Thickness t = 228 mm using E.I.R. model [14][15]. Results using Zahoor et al. [12] also shown as Dashed Lines.

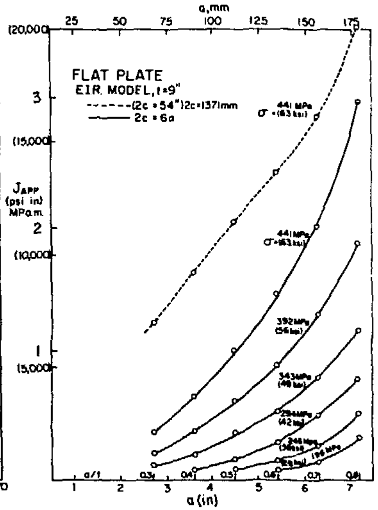

Fig. 5 - Dependence of Japp on Crack Depth a for Various Membrane Stresses σ and Crack Length 2c = 6a in a Flat Plate. One Curve for 2c = 1371 mm also Shown.

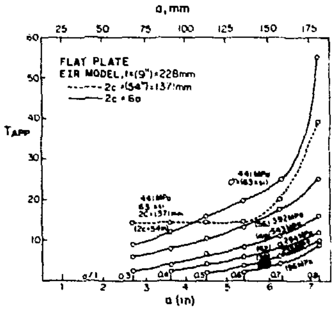

Fig. 6 - Dependence of Tapp on Crack Depth a for Various Membrane Stresses σ and Crack Length 2c = 6a in a Flat Plate. E.I.R. Model [14][15].

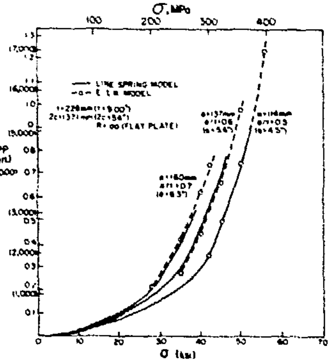

Fig. 7 - Variation of Japp with Membrane Stress σ for Various Crack Depths a using the Line-Spring Model. E.I.R. Model Results included for Comparison. Crack Length 2c = 1371 mm. Flat Plate Thickness t = 228 mm.

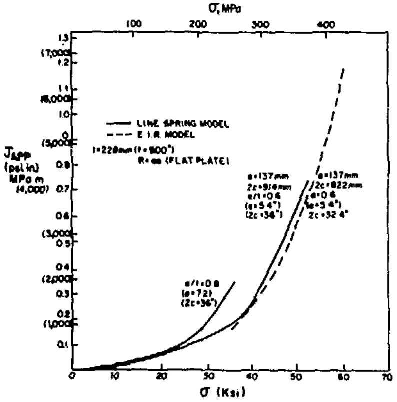

Fig. 8 - Same as Fig. 7 with Crack Length 2c = 914 mm. One Curve Using E.I.R. Model for 2c = 822 mm.

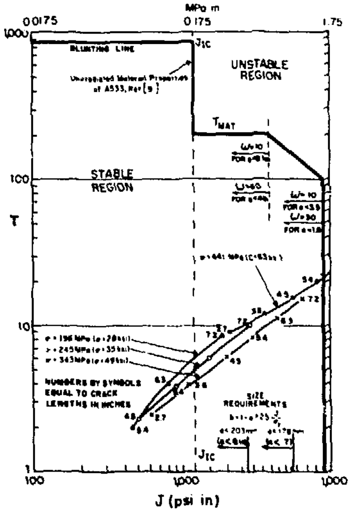

Fig. 9 - Dependence of Tapp on Japp and Limits of Stable T-J Region. Size and ω Requirements also Shown. Data Plotted for Semi-elliptical Family of Cracks (2c=6a), in 228 mm Thick Flat Plates using E.I.R. model.

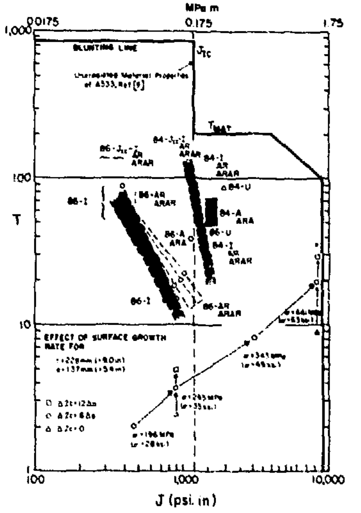

Fig. 10 - Dependence of Tapp on Japp for a 137 mm Deep Crack. Curves also Shown Effect of Increasing Stress Level from 196 MPa to 441 MPa. Effect of Surface Rate [illegible] is Shown at Two Stress Levels. The Stability of the Crack is Analyzed in Reference to [illegible] Properties of Irradiated A533-B Welds from [illegible] et al. [21]

Dup

TWO-PHASE JET LOADS

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ABSTRACT

Two-phase jets are currently being studied to improve engineering models for the prediction of loads on pipes and structures during LOCAs. Multi-dimensional computer codes such as BEACON/MOD2, CSQ, and TRAC-P1A are being employed to predict flow characteristics and flow-structure loading. Our ultimate goal is to develop a new approximate engineering model which is superior to the F. J. Moody design model. Computer results are compared with data obtained from foreign sources, and a technique for using the TRAC-P1A vessel component as a containment model is presented. In general, good agreement with the data is obtained for saturated stagnation conditions; however, difficulties are encountered for subcooled stagnation conditions, possibly due to nucleation delay and non-equilibrium effects.

INTRODUCTION

A nuclear power plant system must be designed in a way that ensures that the consequences of a pipe break will be mitigated. Snubbers and other forms of pipe restraints must be installed to prevent large pipe displacements resulting from thrust forces, and jet deflectors must be engineered to preclude additional effluent damage to the system. At the same time, overdesign of the system must be avoided due to safety and economic considerations. Currently, the NRC uses the Moody model(1) for making two-phase jet load calculations. This model is limited to instantaneous circumferential breaks and may be overly conservative in many cases. The two-phase jet load program at Sandia National Labs is in the process of developing a new approximate engineering model to characterize two-phase jets emanating from breaks in a typical PWR piping system.

To achieve our objective, a two-path program is being followed. The first path is endeavoring to develop an analytic two-phase jet model starting from first principles and making the least number of approximations. Moody's model for jet loads is used as a starting point for this approach. The second path uses large currently existing computer programs to generate predictions for a wide variety of full-scale PWR piping breaks. These predictions will serve as a basis for an engineering model which will supplement and substantiate the analytic expression developed and, in addition, will provide more specific and accurate predictions for certain cases. To increase the reliability of the predictions the computer codes are being verified against experimental data over a wide range of break sizes, small to nearly full-scale.

Most of the work performed to date has focused on the second approach. Three computer codes have been evaluated for modelling a complete two-phase jet blowdown facility. These are CSQ,(2) BEACON/MOD2,(3) and TRAC-P1A.(4) None of these codes were explicitly designed to do the entire two-phase jet problem: modelling pressure vessels, complex pipe networks and valves, critical flow nozzles with possible non-equilibrium fluid states, and multi-dimensional containment regions with impingement targets of arbitrary shape. Of the three codes listed, TRAC-P1A calculations are in best agreement with the experimental data for an overall two-phase jet system.(5) However, other two-dimensional codes such as BEACON/MOD2 or CSQ might produce better axisymmetric jet results than TRAC-P1A if driven with appropriate input boundary conditions, such as those obtainable from RELAP4(6) with a best estimate break flow model (Henry-Fauske/Homogeneous Equilibrium Model). More complex break geometries (longitudinal or double-ended breaks) may still require the use of a full three-dimensional code such as TRAC-P1A or K-FIX.(7)

This paper will briefly discuss some of our results in the following areas:

- steady-state two-phase jet comparisons for saturated stagnation conditions (Kraftwerk Union, Federal Republic of Germany - FRG)
- transient two-phase jet comparisons for subcooled stagnation conditions (Battelle-Frankfurt, FRG)
- effect of pipe break parameters on pipe thrusts and the two-phase jet.

A more complete discussion on these topics can be found in reference 5.

1. Steady-State Two-Phase Jet Comparisons for Saturated Stagnation Conditions

The two-phase jet blowdown test facility operated by Kraftwerk Union, FRG, (KWU), has provided steady-state jet impingement data from saturated stagnation conditions over a variety of orifice diameters (.01-.065 m), stagnation pressures (3.0-10.0 MPa), and orifice to target separations (0.5 D to 10.0 D).(8) Blowdowns were initiated by means of quick opening valves.

A typical comparison between TRAC-P1A results and the KWU data is given in Table 1. Steady-state pipe thrusts were calculated using the relationship given by Moody:(1)

$$F_B = (P_e - P_a)A_e + \rho_e V_e^2 A_e$$

where P is the pressure, ρ is the density, v is the fluid velocity, and A is the pipe exit area (subscripts e and a represent exit and ambient conditions, respectively).

Table 1: TRAC-P1A Comparisons For Kraftwerk Union Data

Test	Pipe Diameter	Pressure	Breakflow (kg/s)			Steady-State Recoil Force (N)		
NW50-6	.05 m	9.62 MPa	Exp.	TRAC	Moody	Exp.	TRAC	Moody (Homogeneous)
			56.4	63.6	76.6	15428	19070	23600
				58*			17400*	20000**

* With annular friction model
**With friction FL/D = 0.81

For the conditions of this test, the TRAC-P1A results are in better agreement with the experimental data than those of the Moody model. Also, the importance of correctly modelling the system friction is apparent.

2. Transient Two-Phase Jet Comparisons for Subcooled Stagnation Conditions

Transient two-phase jet data have been provided by the Battelle-Frankfurt research project RS-50.(9) Subcooled stagnation blowdowns were performed with an exit nozzle diameter of .10 m and an orifice to target separation of .24 m. Test RS-50-C12 (two pressure vessels used to model the BIBLIS A German PWR) was chosen for transient jet analysis because it had the least amount of asymmetric effects caused by the incomplete rupturing of the orifice diaphragm. This test had an initial subcooling of 43 degrees kelvin.

Models used in simulating test RS-50-C12 are shown in Figures 1-3. Figure 1 shows the TRAC-P1A one-dimensional RS-50-C12 model. The separate pressure vessels have been modelled with accumulators, and the system plumbing has been modelled with one-dimensional pipes and a tee component (TRAC-P1A models complex systems with a combination of elementary component models linked together through junctions). Figure 2 shows a unique three-dimensional TRAC-P1A model for RS-50-C12. In this model a vessel component is used to model the separate pressure vessels, the exit piping, and the containment with an impingement target.(5) To construct this model, all of the vessel internals were removed, and appropriate flow areas were set to zero. Blowdown effluent was allowed out of the vessel system through a set of pipes connected to breaks (uppermost axial section). In total, the vessel component contained 20 axial sections, 7 rings, and 2 angular sections. This model is unique to the Sandia Labs program and may be very useful for modelling more complex break geometries, such as longitudinal or double-ended breaks. Figure 3 shows the five volume RELAP4/MOD6 model used to simulate RS-50-C12. A best estimate break flow model was used in the RELAP4/MOD6 calculation (Henry-Fauske/Homogeneous Equilibrium Model--HF/HEM--with a cross over quality, x, of .02). Pseudo steady state loads were calculated from orifice conditions using the thrust relationship previously described. Figures 4 and 5 show comparisons for break flow and jet force, respectively. Best overall agreement with the data was obtained with the RELAP4/MOD6 model (the 3-D TRAC-P1A results are the best for times less than .04 seconds, but for later times the agreement is the worst. This disagreement may have been caused by geometric effects in the 3-D model). The 1-D TRAC-P1A fully implicit model agreed trend-wise with the experimental data, but the values obtained were significantly low (especially in the break flow calculation). The anomalously low 1-D break flow has been attributed to rapid evaporation in the TRAC-P1A calculation near the pipe exit, forcing the flow to equilibrium conditions (no nucleation delay). A future version of TRAC which incorporates a critical flow model and improved constitutive equations could alleviate this problem.

3. Effect of Pipe Break Parameters on Pipe Thrust and Two-Phase Jet Characteristics

To determine the effect of break parameters on pipe thrusts and impingement loads for RS-50-C12, orifice conditions generated by TRAC and RELAP at .02 seconds (time of the first maximum seen in the experimental data) were used as steady-state input boundary conditions to the computer codes SOLA-DF(10) and BEACON/MOD2. Table 2 summarizes the input parameters for the various models.

Table 2: Exit Conditions for Battelle-Frankfurt Test RS-50-C12 at .02 Seconds

Model	Pressure (MPa)	T Saturation	$T_L - T_{sat}$	α	Velocity (m/s)	Mass Flow (kg/s)
HF/HEM	7.53	563.9	-24.3	0.0	64.8	371.
HEM	8.61	573.3	- 8.4	0.0	52.7	304.
MOODY	7.44	563.1	0.0	.103	59.9	311.
TRAC FI 1-D	6.25	551.4	0.0	.5139	119.4	360.
TRAC SI 1-D	5.09	538.2	0.0	.6780	170.1	359
TRAC 3-D	5.92	547.8	8.8	.4729	169.2	545

A wide range of parameters is clearly evident, with TRAC consistently producing the largest exit void fractions (α), the smallest exit pressures, and the largest fluid exit velocities.

Pseudo steady-state orifice forces calculated with various models are compared in Figure 6. With the exception of the Moody break flow model (which may have been applied to a region outside its range of applicability-subcooled stagnation conditions), all of the models produced thrusts of similar magnitude. This agreement between the models was the result of offsetting parameter effects: models which had low exit pressure thrusts $(P_e - P_a)A_e$ also had high mass ejection thrusts, $(\rho_e V_e^2 A_e)$ so that the total thrust, which is the sum of the pressure thrust and the mass ejection thrust, was about the same for all cases.

Figure 7 shows a plate static pressure comparison made for the various break flow models with the computer program SOLA-DF (results with the BEACON/MOD2 code were similar to those shown except for the magnitude of the pressure calculated. This difference arose because no attempt was made to tune the computer results to the experimental data through the use of other input to the codes not directly associated with the input boundary condition: for example, the recommended evaporation rate multiplier was used in the study although better agreement with the data could be obtained with another value). With the exception of the TRAC-3-D model, the other break flow models predicted static pressures which were similar in magnitude. Again, this agreement was caused by offsetting effects in the orifice parameters. Models which had low exit void fractions also had small exit velocities. The larger the initial void fraction, the smaller the resultant plate pressure, and the smaller the initial velocity, the smaller the plate pressure.

No conclusions can be drawn concerning which of the models best fits the experimental data because there may be other input parameters that significantly affect the calculations. Currently we are attempting to establish a best estimate two-phase jet model based on experimental data in order to further rank the break flow models and to provide error bounds for those models which are in disagreement with the best estimate.

4. Conclusions

Based on the work performed to date, the following conclusions can be made:

- for saturated stagnation conditions, TRAC-P1A impingement loads are in better agreement with the experimental data than those of the Moody model
- for subcooled stagnation conditions, no one computer program adequately predicts all aspects of a two-phase jet blowdown. From an overall point-of-view, the calculations performed with TRAC-P1A are in best agreement with the data.
- axisymmetric jets can be predicted with two-dimensional computer codes if orifice parameters are used as input boundary conditions. This technique can be very useful for establishing best estimate models and calculational error bounds.
- steady-state pipe thrusts are not very sensitive to break flow models because of offsetting effects in pressure and mass ejection thrusts.
- static plate pressures for TRAC-P1A break flows and RELAP4/MOD6 (HF/HEM, HEM, and Moody break flow models) are similar in magnitude due to offsetting effects in the orifice parameters (α, V, and P).

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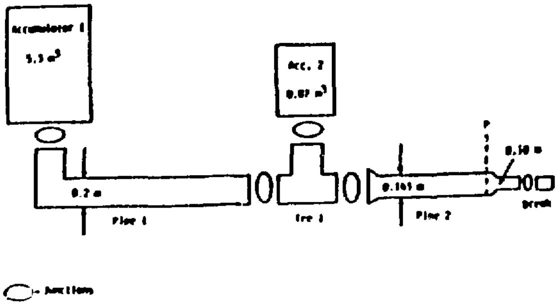
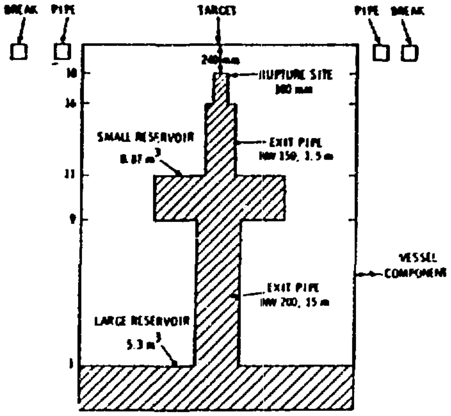
FIGURE 1



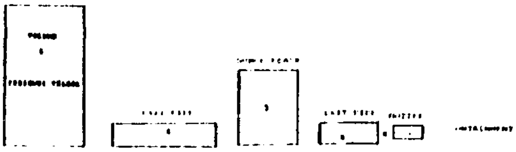
TRAC-PIA 1-D MODEL FOR BATTELLE-FRANKFURT TEST RS-50-C12

FIGURE 2



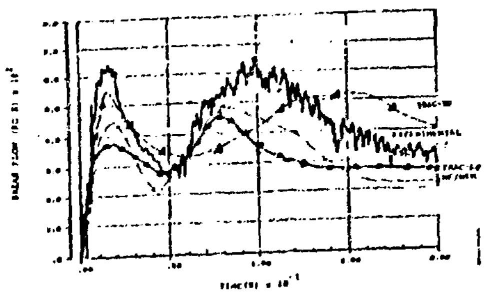
TRAC-PIA 3-D MODEL FOR BATTELLE-FRANKFURT TEST RS-50-C12

FIGURE 3



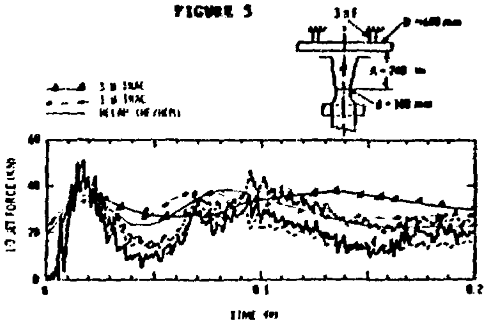
RELAP4/MOD6 5 VOLUME MODEL FOR BATTELLE-FRANKFURT TEST RS-50-C12

FIGURE 4



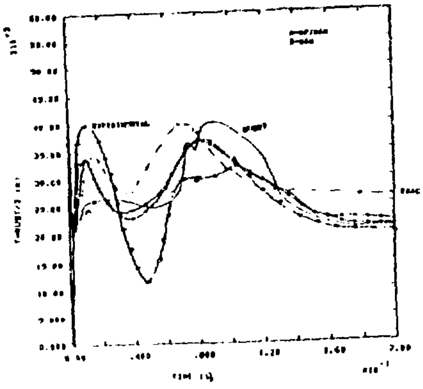
BREAK FLOW COMPARISONS FOR BATTELLE-FRANKFURT TEST RS-50-C12

FIGURE 5

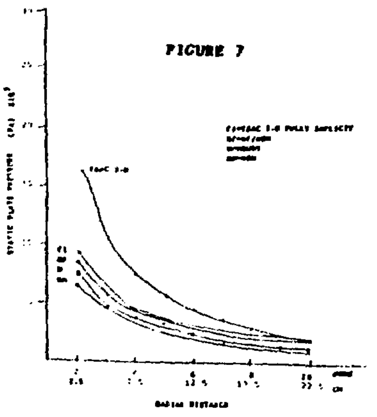


LOAD COMPARISONS FOR BATTELLE-FRANKFURT TEST RS-50-C12

FIGURE 6



THRUST COMPARISONS FOR BATTELLE-FRANKFURT TEST RS-50-C12



SOLA-DF PRESSURE COMPARISONS FOR VARIOUS BREAK FLOW MODELS

RESEARCH NEEDS FOR RESOLVING THE SIGNIFICANT PROBLEMS OF LIGHT WATER REACTOR PIPING SYSTEMS

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ABSTRACT

Past surveys of Light Water Reactor (LWR) piping system problems and recent Licensee Event Report (LER) summaries are studied to identify the significant problems of LWR piping systems and their primary causes. Pipe cracking is identified as the most recurring problem. The most significant cause of pipe cracking, and of other piping system problems in general, is determined to be the vibration of pipes due to operating pump-pipe resonance, fluid-flow fluctuations, and vibration of pipe supports. Recent, ongoing, and planned research in the United States relevant to the identified piping system problems is evaluated, and, on this basis, the need for further research toward resolving these problems is determined. Programs to carry out needed research are suggested on topics related to pipe vibration, thermal and dynamic pipe behavior, piping analysis, fatigue, design loads, and the problem of radiation buildup in corrosion products.

INTRODUCTION

Problems with pipes and pipe fittings are responsible for about 10% of all safety-related events [1] and about 7% of all outage time [2] at LWRs. Identification and resolution of these problems can lead to increased overall plant safety and significant economic benefits.

The objectives of the study reported in this paper were to identify the significant problems of LWR piping systems, to determine the research needs for resolving these problems, and to recommend programs to carry out the needed research. LER summaries from the U.S. Nuclear Regulatory Commission (NRC) data base were used as the main source of information for identification and characterization of the problem areas; findings of previously conducted surveys of piping failures [2, 3, 4, 5, 6, 7] were also used. The identified problems and their causes were analyzed, and the research areas relevant to the resolution of the problems were delineated. Then recently completed, ongoing, and planned research in these areas were evaluated to determine the areas most in need of further research. These areas were ranked, and programs on specific research topics were suggested.

PROBLEM IDENTIFICATION

To identify and characterize the significant problems of LWR piping systems, a survey of reportable occurrences involving pipes and fittings was conducted.

(A reportable occurrence is an unscheduled incident or event that the NRC determines is significant from the standpoint of public health or safety; Licensee Event Report, or LER, is a generic term for reportable occurrence.) For the survey, LER summaries submitted to the NRC data base during the period June 1, 1976, through May 16, 1979, were used. LER data submitted prior to this period have been extensively studied by previous surveys [2]. The survey covered a total of 354 reportable occurrences. Regardless of the number of problems reported in a single LER summary, each summary was counted as one reportable occurrence.

Some of the more important findings of the survey are summarized below:

- Cracking of pipes and fittings was the most significant problem, constituting 52.0% of the reportable occurrences. Of the cracks reported, 77.2% (40.1% of total) were through-the-wall cracks and were discovered by observed leaks. Causes of cracking were cited as follows:

Cause	Percent of Reportable Occurrences Involving Cracks
Mechanical vibration	34.8
Weld defects	12.0
Stress corrosion	9.2
Thermal fatigue	6.5
Installation error	6.0
Inadequate support	6.0
Material flaws	3.3

- The most cited causes of reportable occurrences involving all types of failure, including cracking, were as follows:

Cause	Percent of All Reportable Occurrences
Mechanical vibration	22.3
Weld defects	8.5
Personnel error	6.8
Erosion and cavitation	6.2
Installation error	5.9
Stress corrosion	4.8
Corrosion	3.7
Thermal fatigue	3.4

- Of the reported failures, 31.6% occurred at welds and 9.6% at weld-heat-affected zones of pipes, for a combined total of 41.2%

Because the survey identifies cracking as the most significant problem of LWR piping systems, there is a need for understanding and preventing the cracking phenomenon. Furthermore, because most of the cracks reported in the survey were discovered through observed leaks, there is also a need for discovering cracks before they cause a rupture.

Other significant problems identified by the survey are related to erosion and cavitation, general corrosion, and analysis and design of piping systems.

IDENTIFICATION OF RESEARCH NEEDS

The problems identified as being significant were analyzed as to their causes, their possible consequences, and the research areas relevant to their resolution. The following is a list of the significant piping problems and the research areas relevant to their resolution.

Problem	Relevant Research Area
Pipe cracking	Flaw detection and evaluation Fracture mechanics Fatigue studies Stress corrosion cracking Postulated pipe rupture Pipe vibrations Welding technology Thermal versus dynamic pipe behavior
Erosion and cavitation	Erosion and cavitation
Corrosion-related problems	Radiation buildup in corrosion products
Design and analysis problems	Pipe design loads Piping analysis

A literature survey was conducted to evaluate the recently completed, ongoing, and planned research efforts in each of the research areas listed above and to determine the state of knowledge in these areas. On the basis of this survey, research needs in each area and specific research topics relevant to resolution of the piping problems were identified.

To properly allocate future funds to programs directed toward resolving the significant problems of LWR piping systems -- that is, to allocate funds in such a way that benefits gained from such research in terms of increased safety and reduced costs are maximized -- a rational approach for assigning priorities to the identified research area is needed. Such an approach would involve determination of the state of knowledge and the needed research in a specialized area and thus would ideally require a panel of experts in that area. Also, a rational approach would require quantifying concepts such as significance and safety. Because both of these requirements were beyond the scope of the study being reported herein, the study's findings, which were made using the best available information and some subjective judgment, should be considered only as general guidelines to future research plans.

In an attempt to rank or assign priorities to the research areas under consideration, an evaluation matrix was developed and is shown in Table I. In this table, the piping problems and research areas that were evaluated and ranked are listed in the first column. Columns 2 through 13 list the factors that were considered for determining the relative significance of the research areas. The priorities assigned to the areas and shown in Column 2 reflect the need for further research as determined through the literature survey. If the survey indicated a large demand for further research in an area and the present level of effort was not considered to be sufficient, then the research area was assigned a priority of one. If the survey indicated that there was little need for further research or that the present and planned level of effort in the industry was adequate, a lower priority (two or three) was assigned to the research area.

Columns 3, 4, and 5 of the matrix reflect the expected relative time and cost of the program. If the expected time or cost was very small, a zero was assigned. If the program was expected to take too long to complete or to cost too much, a number between zero and minus three was assigned.

Columns 6 through 12 reflect the relative benefits that might be gained from the program. A zero indicates minimal benefit, whereas a three indicates a high level of benefit.

The numbers given in Columns 3 through 12 of each row of the matrix were summed algebraically in Column 13. On the basis of these sums, the research areas were reordered and ranked as shown in Table II. The assumption made in using the sums in Column 13 of the matrix as the basis for ranking the research areas was that all the factors considered are of equal weight. It was determined that the highest priority research areas relevant to the problems of LWR piping systems are, in order of significance:

1. Pipe vibration
2. Thermal versus dynamic pipe behavior
3. Piping analysis
4. Fatigue studies
5. Piping system design loads
6. Radiation buildup in corrosion products

RECOMMENDED PROGRAMS

Programs to carry out the needed research in the highest priority areas relevant to the resolution of the significant problems of LWR piping systems are briefly described below.

Pipe Vibration

Objective:		To reduce or eliminate pipe cracking due to mechanical vibrations caused by pump-pipe resonance, fluid flow fluctuations, or support vibrations.
Suggested Topics:	1.	Operational data collection: Instrumentation of various piping systems in operating LWR plants to collect vibration data and to characterize vibration problems.
	2.	Analytical developments: Development of analytical techniques to analyze piping systems for predicting potential vibration problems.
	3.	Investigation of possible use of snubbers in vibration control.
	4.	Development of design guidelines to account for mechanical vibrations.

Thermal versus Dynamic Pipe Behavior

Objective:		To resolve the inherent conflict between piping designs for thermal and dynamic loads (e.g., earthquakes).
Suggested Topics:	1.	Thermal and dynamic pipe behavior monitoring: Instrumentation of various piping systems in operating LWR plants located in high seismicity areas; collection of data on thermal behavior of piping systems,

performance of forced vibration tests, and possible measurement of seismic response in the event of an earthquake.

2. Analytical studies: Evaluation of operational data and development of alternate approaches to the design of piping systems.
3. Snubber-support development: Development of alternate devices or methods to replace or improve snubbers.

Piping Analysis

Objective: Development of analytical capabilities to more accurately predict piping system response and safety margins under all service and accident conditions.

Suggested Topics:

1. Development of efficient nonlinear piping analysis codes.
2. Development of realistic approximate seismic analysis techniques, taking into account the multiple support seismic input motions.

Fatigue Studies

Objective: To study the high- and low-cycle fatigue properties of LWR piping steels and weldments.

Suggested Topics:

1. High-cycle fatigue studies: Performance of realistic tests of piping systems.
2. Low-cycle fatigue studies: Performance of realistic tests of piping systems.

Piping System Design Loads

Objective: To better define LWR piping system design loads.

Suggested Topics:

1. Collection and evaluation of recent data relevant to piping loads: Development of more realistic design load definitions.
2. Pipe rupture criteria revision: Development of more realistic, rational pipe rupture criteria.
3. Seismic loads: Evaluation of current seismic design loads and assumptions.
4. Load combinations: Development of rational, probabilistic methods for combining design loads.

Radiation Buildup in Corrosion Products

Objective: To understand the mechanism for crud growth and radioisotope incorporation into this crud in stainless steel LWR piping; to develop effective methods for decontamination of BWR primary systems.

Suggested Topics:

1. Surveys of radiation buildup: Radiation field mapping of operating BWR plant primary piping.

2. Analytical models: Development of analytical models for predicting future radiation levels.
3. Decontamination processes: Development and testing of processes for decontamination of BWR primary systems.

REMARKS

Findings, conclusions, and recommendations presented in this paper should be treated as general guidelines for planning future research efforts toward improving the safety and performance of LWR piping systems. The study summarized herein was very general in scope and by intent and necessity was not detailed and precise. Detailed scoping studies on the suggested research topics would be recommended for planning future research.

ACKNOWLEDGMENTS

This paper is based on a project titled "Identification of Significant Problems Related to Light Water Reactor Piping Systems." The project is part of the Light Water Reactory Safety Technology Program within the Nuclear Power Development Division of the U.S. Department of Energy (DOE) and is administered by Sandia Laboratories, Albuquerque, New Mexico, as Manager of the DOE-LWR Safety Technology Program.

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TABLE I

Evaluation of Research Areas

					Positive Impact On							
Research Area (1)	Priority (2)	R/D Time Requirements (3)	Cost of R/D (4)	Cost of Implementing Results (5)	Safety (6)	Construction Costs (7)	O/M Costs (8)	Plant Availability (9)	Safety Margins (10)	Licensing Time (11)	Generic Applicability (12)	Sums (13)
Pipe cracking:												
Flaw detection and evaluation	3	-2	-2	-1	3	0	3	3	3	1	3	11
Fracture mechanics	2	-2	-2	0	2	0	3	3	3	0	3	10
Fatigue studies	1	-2	-2	0	2	0	3	3	3	0	2	9
Stress corrosion cracking	3	-3	-2	-2	2	0	2	3	3	1	1	5
Postulated pipe rupture	2	-1	-2	0	2	3	2	1	3	2	2	12
Pipe vibration	1	-1	-1	0	2	1	3	3	2	1	2	12
Welding technology	2	-2	-2	0	2	1	3	3	2	0	2	9
Thermal versus dynamic pipe behavior	1	-1	-1	-1	1	1	3	3	2	1	2	10
Erosion and cavitation	2	0	0	-1	2	0	2	2	1	0	2	8
General corrosion: Radiation buildup in corrosion products	1	-2	-1	-1	2	1	3	3	0	0	3	8
Design/analysis problems:												
Piping system design loads	1	-3	-2	0	2	3	1	1	3	2	2	9
Piping analysis	1	-2	-1	-1	2	1	2	2	3	2	2	10

TABLE II

Ranking of Research Areas

Priority	Sum[a]	Research Area
1	12	Pipe vibration
2	12	Postulated pipe rupture
3	11	Flaw detection and evaluation
1	10	Thermal versus dynamic pipe behavior
1	10	Piping analysis
2	10	Fracture mechanics
1	9	Fatigue studies
1	9	Piping system design loads
2	9	Welding technology
1	8	Radiation buildup in corrosion products
2	8	Erosion and cavitation
3	5	Stress corrosion cracking

[a] From Column 13, Table I

CRUSH PIPE RUPTURE RESTRAINT DESIGN

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ABSTRACT

Crush pipe rupture restraints are utilized at Duke Power Company for control of the movement and forces of high energy process pipe after postulated ruptures. Duke Power sponsored a test program to define crush pipe characteristics for design application. In rupture restraint design, simplified hand calculations may be used to obtain the blowdown force of the ruptured process pipe, the deflection of the crush pipe, and the load on the restraint structure. Alternatively, a time history analysis of the blowdown and a rigorous elastic-plastic computer calculation of deflection and load is sometimes utilized. A comparison between crush pipes and the two most commonly used energy absorbers - crush pads and U-rods - shows that crush pipes have the advantages of reduced material cost and less procurement problems.

INTRODUCTION

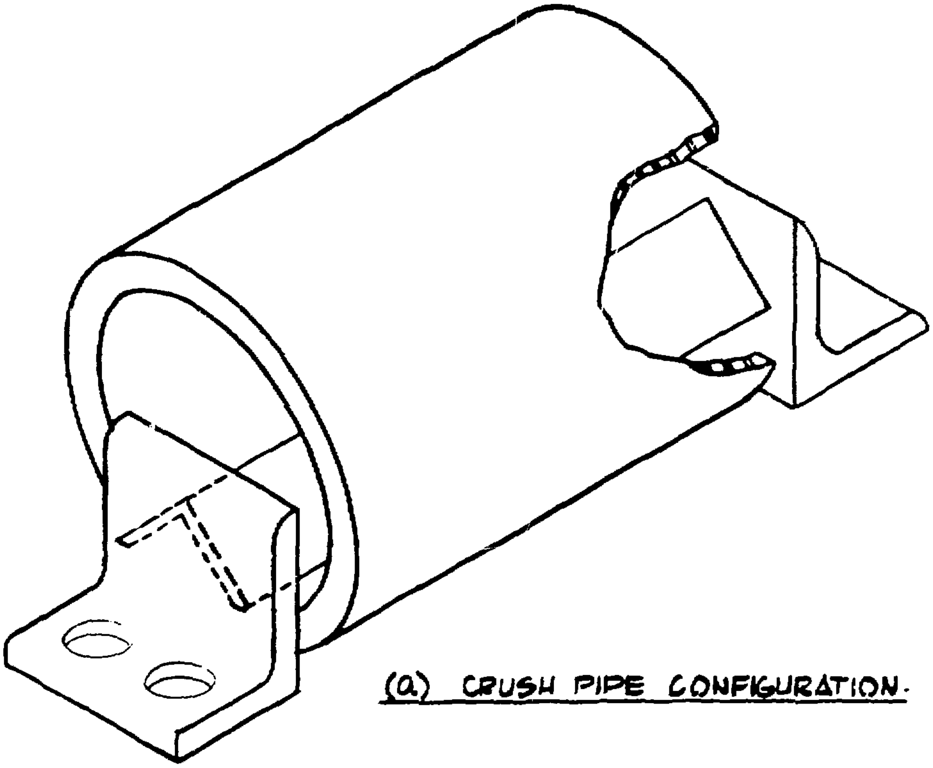
The requirement for pipe rupture restraints is a direct result of General Design Criterion (GDC) 4, "Environmental and Missile Design Basis", of Appendix A to 10CFR50. GDC 4 states, in part, that "Structures, systems, and components important to safety shall be designed to accommodate the effects of . . . . . . postulated accidents, including loss of coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of . . . . pipe whipping and discharging fluids . . . . ."[1] Standard Review Plans 3.6.1 and 3.6.2 and Regulatory Guide 1.46 provide guidance for rupture postulation and protection.

The function of an energy absorber rupture restraint is to (1) control the movement of a ruptured process pipe and (2) minimize the forces on the building structure by absorbing the energy of the whipping pipe. The rupture restraint typically must not contact the process pipe during any normal or abnormal operating conditions, except in the postulated event of pipe rupture. This paper describes the application of crush pipes as energy absorbers in rupture restraint design at Duke Power Company. Figure 1 shows a simplified typical configuration for a crush pipe rupture restraint.

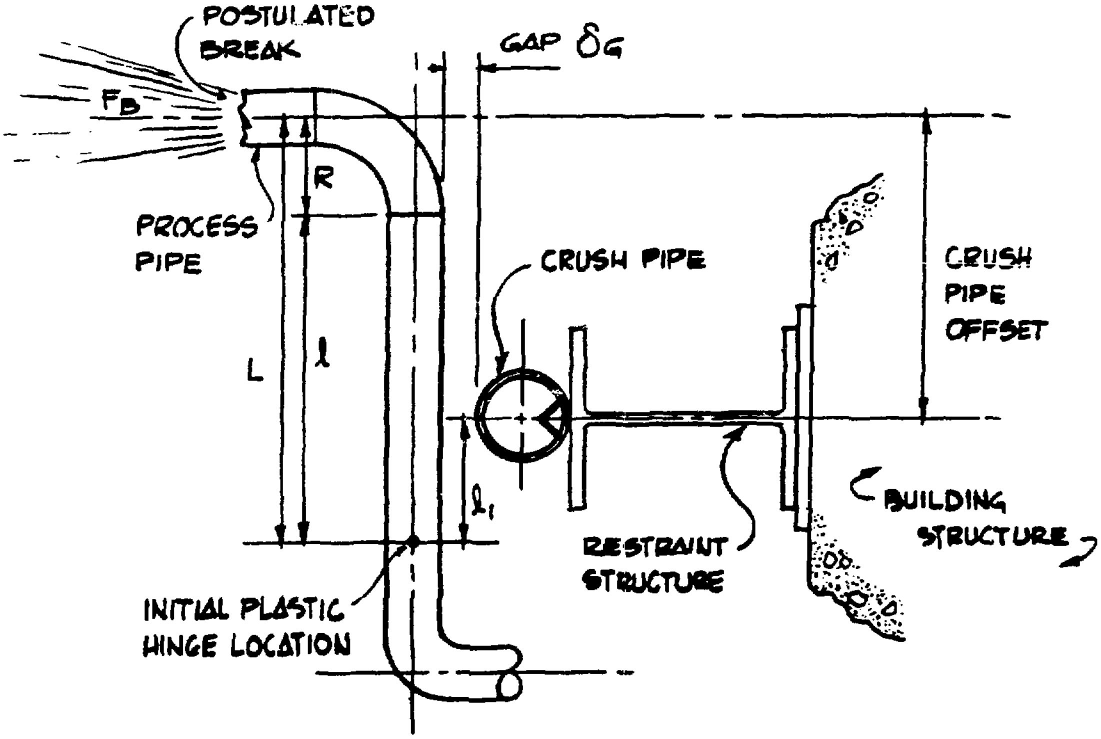
DESIGN METHODS FOR CRUSH PIPE RUPTURE RESTRAINTS

The design of rupture restraints is a two step process involving the computation of the rupture blowdown force and the application of that force to the crush pipe and rupture restraint structure. Simple hand calculations or more complicated computerized methods may be utilized. The simplified techniques are first described in this section, and then more sophisticated computer methods are presented. The simplified method of calculating the blowdown force of the ruptured process pipe utilizes the following formula:

$$F_B = C_T^M P_o A \qquad (1)$$



(a) CRUSH PIPE CONFIGURATION.



(b) RUPTURE RESTRAINT & PROCESS PIPE ARRANGEMENT

FIGURE 1.

where F_B = steady state blowdown force, lb.

C_T^M = maximum steady state thrust coefficient, (unitless)

P_o = system stagnation pressure, lb./in.^{2}

A = break area of the pipe, in.^{2}

This steady state force is assumed to be reached instantaneously at the time of rupture and to remain constant thereafter. Loss of pressure head due to friction forces is ignored. Figure 2 illustrates the relationship between a rigorous time history analysis and this simplified approach for calculating blowdown force. For rupture restraint design, typically $C_T^M = 1.26$ is used for saturated water, water-steam mixtures, and superheated steam. For sub-cooled non-flashing water, $C_T^M = 2.0$ may be used. These conservative values of C_T^M along with a conservative dynamic amplification factor result in an overestimate of structural load. The overestimate is further increased in the presence of limited source resevoirs, flow restrictors, and check valves in the piping system.

The second step in crush pipe rupture restraint design is the computation of the crush pipe deformation and the load on the restraint back-up structure. The simple energy balance method as described below is often used with the simplified blowdown force calculated by Eq. (1).

For a crush pipe, the force versus deformation characteristics are defined by the bi-linear force-deformation curve[2] depicted in Fig. 3(a). In the figure,

F = resisting force of crush pipe assuming rigid back-up structure, lb.
δ = deformation of crush pipe, in.
F_C = collapse force, or force at initiation of plastic behavior, lb.
δ_C = deformation at initiation of collapse (plastic hinge), in.

The following equations describe the crush pipe force-deformation relationships:

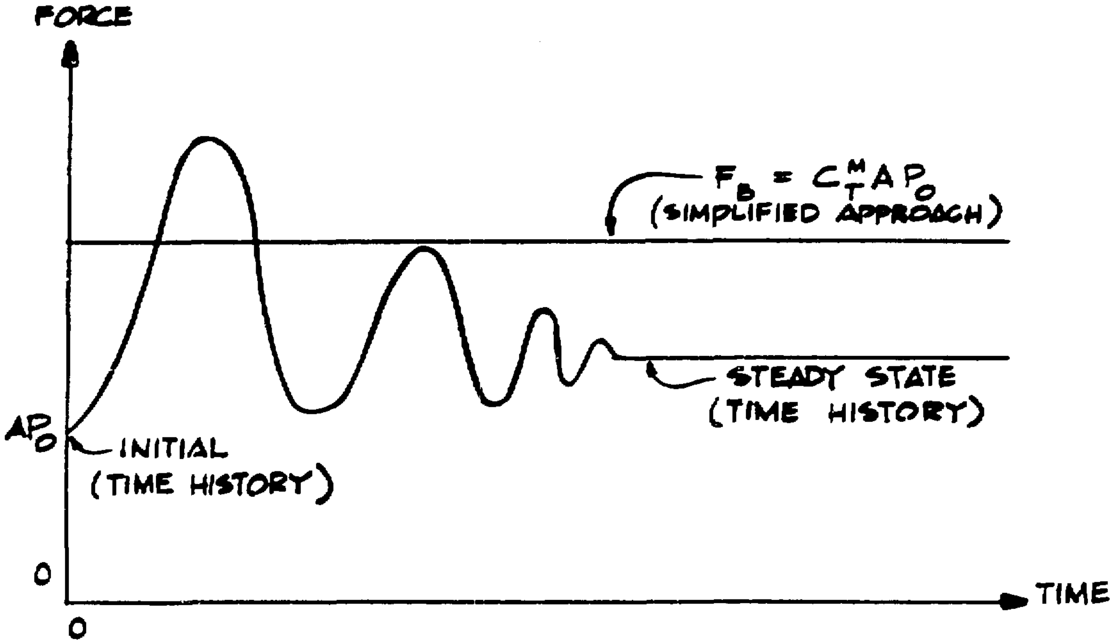
$$F = \begin{cases} K_1\delta & \text{, elastic region} \qquad (2a) \\ K_2\delta + (K_1-K_2)\delta_C & \text{, plastic region} \qquad (2b) \end{cases}$$

where K_1 = elastic region slope for crush pipe force versus deformation curve, lb./in.
K_2 = plastic region slope for crush pipe versus deformation curve, lb./in.

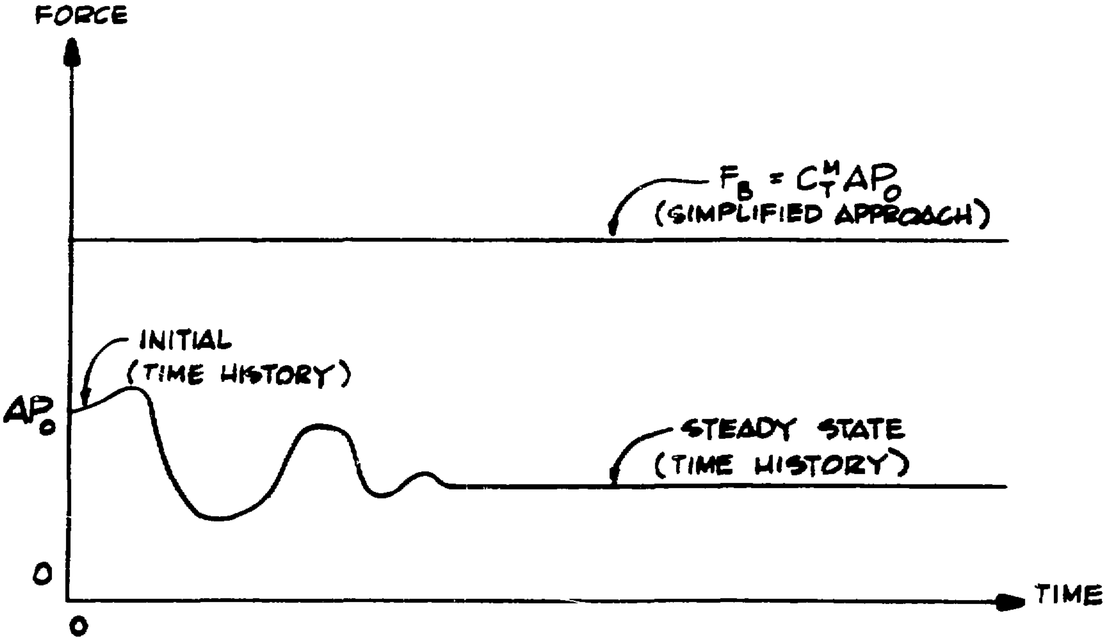
The simplified energy balance method for determination of crush pipe deflection is now applied. The kinetic energy of the whipping pipe, the energy absorbed during formation of the initial plastic hinge (Fig. 1(b)), and the strain energy of the deforming process pipe are conservatively assumed to negate each other. The external energy associated with the blowdown force is absorbed by the crush pipe; hence the energy balance involves the following relationships:

Energy associated with the external blowdown force =
Energy absorbed by crush pipe deformation (3)

or Blowdown force times pipe travel =
Area under force-deflection curve. (4)

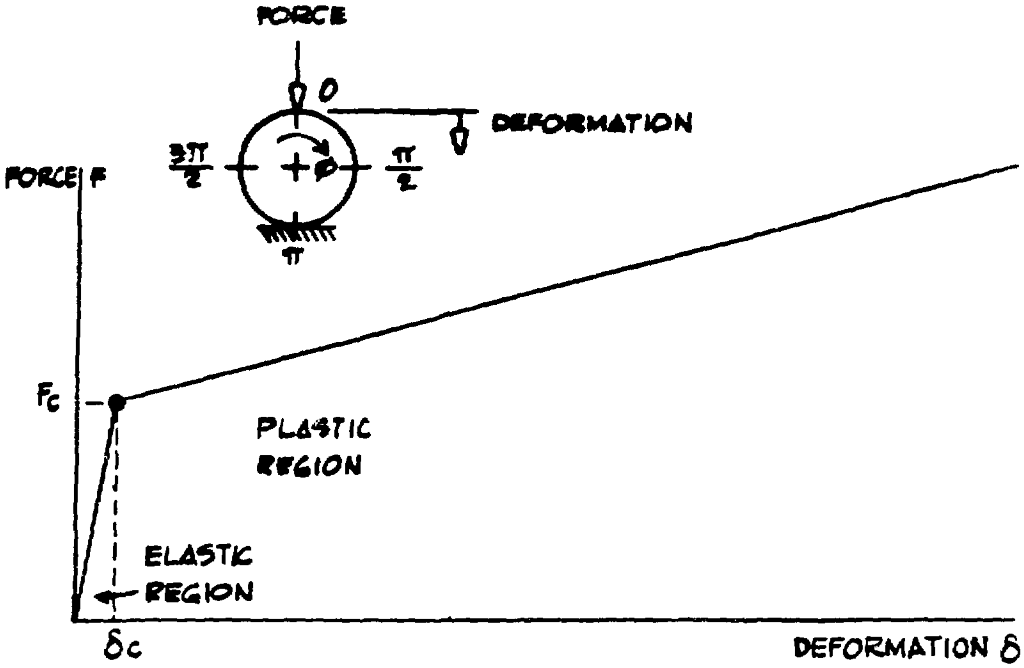


(a) VERY LOW FRICTION ($C_T \geq 1.0$)

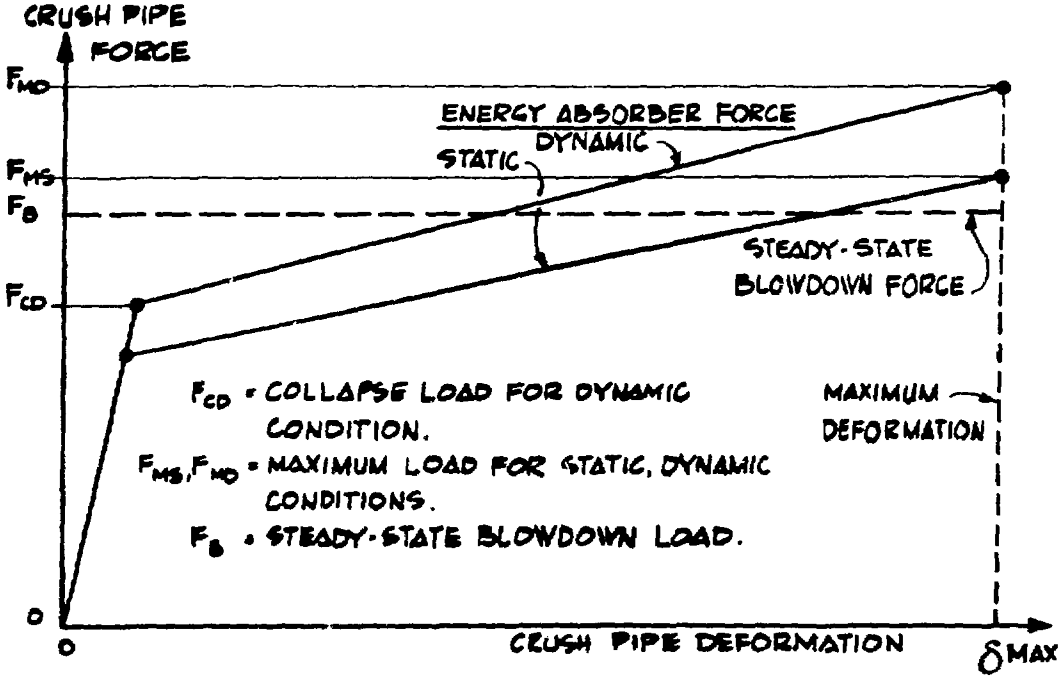


(b) FRICTION ($C_T < 1.0$)

FIGURE 2. PIPE RUPTURE BLOWDOWN FORCE



(a) CRUSH PIPE BI-LINEAR FORCE VERSUS DEFORMATION RELATIONSHIP.



(b) NOMENCLATURE FOR CALCULATION OF RESTRAINT/PLANT STRUCTURE DESIGN LOAD FOR CRUSH PIPE APPLICATION

FIGURE 3.

Assuming no crush pipe offset (Fig. 1(b)), and assuming a rigid back-up structure, the energy balance in equation form becomes ($\delta_M > \delta_C$):

$$1.2\ F_B(\delta_G + \delta_M) = \tfrac{1}{2}K_2\delta_M^2 - (K_2 - K_1)\delta_C\delta_M + \tfrac{1}{2}(K_2 - K_1)\delta_C^2 \qquad (5)$$

where δ_G = distance between process pipe surface and crush pipe surface along line of crush, in.

δ_M = maximum crush pipe deformation, in.

and the remaining symbols are as previously defined. Note that the blowdown force is increased by 20% to account for the effects of rebound as specified in SRP 3.6.2.[4] Deformation of the crush pipe is given by the following general quadratic solution:

$$\delta_M = \frac{-B \pm \sqrt{B^2 - 4AC}}{2A} \qquad (6)$$

where $A = \tfrac{1}{2}K_2$

$B = -(1.2)F_B - (K_2 - K_1)\delta_C$

$C = \tfrac{1}{2}(K_2 - K_1)\delta_C^2 - (1.2)F_B\delta_G$

In order to apply Eq. (6), the elastic and plastic slopes, K_1 and K_2, must be defined at elevated temperatures and under dynamic loading conditions. Duke Power sponsored a test program to define these and other response characteristics of crush pipe.[3] In doing so existing analytical relationships were verified or modified and some new analytical relations were developed. Static and dynamic tests were performed on one diameter long seamless carbon steel pipe (A106 Grade B, with Certified Mill Test Reports). Crush pipe size ranged from 4" Sch 80 to 6" Sch 160. Strain rate, maximum crush pipe deformation and impact velocity were varied over appropriate ranges. A relationship between static and dynamic pipe crush characteristics, and temperature effects on yield stress were evaluated. The resulting relationships are given below:

$$K_1 = 4.48\ El\ \frac{(t/D)^3}{(1-t/D)^3} \text{ for } 1 \leq 2D \qquad (7)$$

$$K_2 = 4.86\ \sigma_y l\ \frac{(t/D)^2}{(1-t/D)} \text{ for } 1 \leq 2D \qquad (8)$$

where E = modulus of elasticity of crush pipe, psi

l = length of crush pipe, in.

D = outside diameter of crush pipe, in.

σ_y = yield stress of energy absorber material for dynamic condition at applicable temperature

t = nominal wall thickness of crush pipe, in.

In Equation (8)

$$\sigma_y = K_{DT}\sigma_{yrs} \qquad (9)$$

where σ_{yrs} = room temperature static yield stress value obtained from Mill Test Report, psi.

K_{DT} = adjustment factor for σ_{yrs} to obtain yield stress at temperatures from 70°F to 650°F under dynamic loading, (unitless).

Then

$$K_{DT} = 1.20 - 0.12 \frac{T_{avg}-70}{580} \qquad (10)$$

where T_{avg} = average temperature of the crush pipe, °F.

The allowable temperature range within which crush pipes are designed to function is 70°F to 650°F.

The test program results as applied in Eqs. (2) through (11) were submitted to the NRC by Duke Power as an "Energy Absorber Design Procedure".[2] Approval of the procedure was received in January, 1978.

An additional refinement to Eq. (5) is appropriate because it contains the inherent assumption that the crush pipe is co-linear with the blowdown line of force. Thus the offset shown in Fig. 1(b) becomes zero, and some type of elbow pad is required to be welded or clamped onto the process pipe to provide a flat impact surface. As a practical matter the crush pipe is usually offset from the blowdown line of force as shown in Fig. 1(b). The moment arm thus created results in an increase in the energy imparted to the crush pipe. When the moment arm effect is included in Eq. (5), it becomes[5]

$$1.2\,F_B(\delta_G + \delta_M)\frac{(\ell + R)}{\ell_1} = \tfrac{1}{2}K_2\delta_M^2 - (K_2 - K_1)\delta_C\delta_M + \tfrac{1}{2}(K_2 - K_1)\delta_C^2 \qquad (12)$$

where
R = elbow radius, in.
ℓ = distance from center of elbow radius to location of initial plastic hinge, in.
ℓ_1 = distance from center of crush pipe to location of initial plastic hinge, in.

The factor $(\ell + R)/\ell_1$ is proportional to the amount of offset between the crush pipe centerline and the blowdown line of force. The distance $\ell + R$, designated L, can be calculated by the following equation:

$$T_i L^2 - 3M_p L - 6M_p \frac{M}{m} = 0 \qquad (13)$$

where
$T_i = P_oA$ = initial blowdown force, lb.
$L = \ell + R$, in.
M_p = plastic moment capacity of straight portion of process pipe, in. lb.
M = concentrated mass of the process pipe from the postulated break through the first elbow, lb.
m = mass/length of straight pipe, lb./in.

If L is greater than the distance between the first and second elbow upstream from the break (Fig. 1), a permanent plastic hinge is assumed to form at the latter. If there is a concentrated mass, e.g., a valve, located between the two elbows, Eq. (13) becomes invalid and must be refined.

Finally,

$$M_p = 4/3\ \sigma_{yp}\ (R_p^3 - r_p^3) \tag{14}$$

where σ_{yp} = minimum yield stress of process pipe at temperature, psi.
R_p = outside radius of process pipe, in.
r_p = inside radius of process pipe, in.

The result of solving Eq. (12) for crush pipe maximum deflection, δ_M, now becomes:

$$\delta_M = \frac{-B \pm \sqrt{B^2 - 4AC}}{2A} \tag{15}$$

and the modified values of the constants A, B, and C are:

$$A = \tfrac{1}{2}K_2$$

$$B = -\frac{1.2F_BL}{\ell_1} - (K_2 - K_1)\delta_c$$

$$C = \tfrac{1}{2}(K_2 - K_1)\delta_c^2 - \frac{1.2F_BL}{\ell_1}\delta_G$$

The crush pipe should be designed to have a static load capability at maximum deflection, δ_M, which exceeds the steady state blowdown force, F_B. This is illustrated in Fig. 3(b), where $F_B < F_{MS}$. Diametrical deformation is limited to an amount not to exceed the lesser of (1) half the outside diameter of the crush pipe or (2) the maximum flattening deformation as prescribed by ASTM A530, δ_{ASTM},

where $$\delta_{ASTM} = \frac{D-t}{1 + 14.29(t/D)}$$

and D = outside diameter of crush pipe, in.
t = nominal wall thickness of crush pipe, in.

Crush pipes are designed using standard pipe sizes with length limited to twice outside diameter.

The rupture restraint backup structure is designed to limit member stresses to 90% of the yield stress. To insure that it is capable of supporting the steady state blowdown force after crush pipe deformation, the backup structure should satisfy the requirements of static equilibrium. Also, it must be capable of withstanding the maximum dynamic load due to impact of the whipping process pipe without loss of integrity. The restraint backup and/or plant structures must be designed for the greater of (1) dynamic collapse load plus dynamic maximum load of the crush pipe or (2) 2.4 times the steady state blowdown load. (See Fig. 3(b).) This can be written as:

$$F_R = \mathrm{MAX}\ \{F_{CD} + F_{MD};\ 2.4F_B\} \tag{16}$$

The 2.4 factor applied to the blowdown force is the combination of a dynamic amplification factor of 2.0 and a 20% increase in blowdown force to account for rebound effects.[5] Equation (16) is known to be conservative when compared to results of rigorous time history analyses.

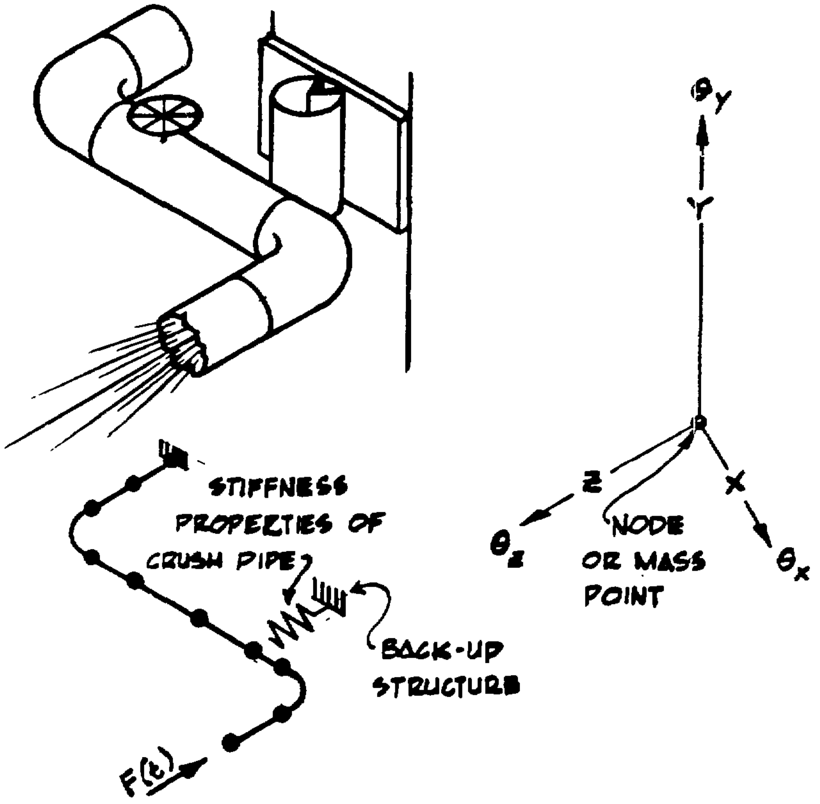
The methods of crush pipe rupture restraint design described previously in this paper result in a conservative, "overdesigned" rupture restraint. In cases where a less conservative design is deemed necessary, a rigorous time history of the ruptured pipe blowdown force may be used in lieu of Eq. (1). Then a non-linear elastic-plastic structural analysis is applied rather than the simple energy balance method. In such cases at Duke Power the computer program PRTHRUST[6], a modification of the NRC accepted LOCA code RELAP, is utilized to obtain the blowdown time history. A crack is assumed to require one milli-second to open. The transient reaction force acting at the break plane is calculated. The blowdown force is traced until steady state is reached or until the energy source is depleted (Fig. 2).

For the rigorous structural analysis, the computer program PIPERUP[6,7] is used. This program will accommodate three dimensional piping systems. The process pipe is mathematically modeled as a series of flexible, weightless, structural members connecting discrete node points (see Fig. 4(a)). Straight runs of process pipe are mathematically represented by straight beam elements, and elbows are represented by curved beam elements. The distributed weight of the process piping and concentrated weights such as valves are lumped at nodes selected as mass points. Both the mass and stiffness damping properties of the process pipe are input. An axial spring is used to represent the crush pipe, and the gap between the rupture restraint and crush pipe is modeled. Properties of the crush pipe are input as previously described (Fig. 3(a)).

The PIPERUP dynamic analysis proceeds in increments, tracing the non-linear elastic-plastic deformation of the process pipe and crush pipe. Stiffness characteristics of the process pipe, illustrated in the tri-linear curve of Figure 4(b), are updated based on the computed response at the end of each time increment. As strain proceeds, the process pipe response is represented by linear elastic behavior, linear strain hardening, and then perfectly plastic behavior. Unloading is assumed to be parallel to the elastic line. Crush pipe stiffness is represented as described previously in Fig. 3(a). Calculation is terminated when equilibrium is established or crush pipe deformation limit is exceeded. Program output includes the maximum reaction force on the crush pipe and forces and moments in each process piping element. If a crush pipe or piping element has gone plastic, this is noted along with the amount of deflection. Thus the conservatisms are reduced by this rigorous design method.

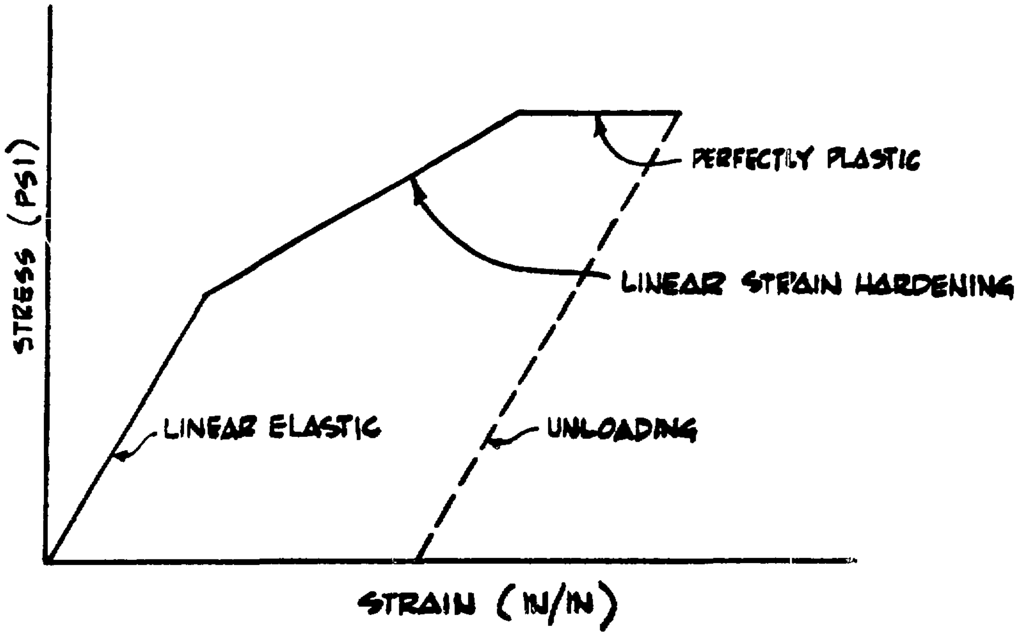
SOME PRACTICAL CONSIDERATIONS IN RUPTURE RESTRAINT DESIGN

A comparison of the PIPERUP computer analysis and the simple hand calculation methods of rupture restraint design is now appropriate. Smaller structural loads and crush pipe sizes result from PIPERUP analyses, reflecting the removal of many of the conservatisms present in the simplified energy balance method. In comparison cases run at Duke Power, the structural load has been reduced by as much as 50% by utilizing the PIPERUP computer analysis. On the other hand, the hand calculation method reduces design time and cost while producing over-designed rupture restraints. Our conclusion is that there is an advantage in using PIPERUP on large diameter lines where extremely large structural loads (on the order of one thousand kips) are involved. Examples are main steam and feedwater lines. For smaller lines with smaller loads, there is no clear



(a) PIPE GEOMETRY AND PIPERUP MODEL

FIGURE 4.



(b) PIPING STRESS-STRAIN CHARACTERISTICS FOR PIPERUP advantage to either the simple or rigorous method. The reduced design time and cost with the simple energy balance method may be negated by the increase in material cost and installation problems in such cases.

The two most commonly used energy absorbers are crush pads and U-rod. A comparison between these and crush pipes is presented in the categories of material cost, design cost, procurement problems, space requirements, and installation problems:

a. Cost of Fabricated Material - Crush pipes show a tremendous advantage over crush pads and U-rods in the category of fabricated material cost. A crush pad or U-rods may cost from 20 to 50 times more than the equivalent crush pipe.
b. Design Cost - The design costs for the three types of energy absorbers would be approximately the same.
c. Procurement Problems - Crush pipes are ordered by the same procedure as all other A106 Grade B pipe in the plant. Crush pads and U-rod restraints require separate specifications and vendor contracts. Thus crush pipes can be procured on short schedules, and design changes resulting in the use of different size pipe have minimal impact. Crush pads require long procurement lead times and provide almost no design flexibility late in plant construction. The same is true to a lesser extent for U-rod restraints.
d. Space Requirements - Crush pads show an advantage where there is a critical space limitation. These energy absorbers have a greater energy absorption capacity per unit volume of occupied space.
e. Installation Problems - No increase in installation problems is seen when using crush pipe rather than U-rod restraints or crush pads. U-rod restraints are the most susceptible to interferences from other plant structures. Crush pads are the most likely to be damaged once installed.

The foregoing comparisons point out the overall advantage which crush pipes have for most energy absorber applications.

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ASSESSMENT OF CURRENT LIGHT WATER REACTOR PIPING DESIGN LOADINGS BASED ON OPERATING EXPERIENCE

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ABSTRACT

This paper will describe the results of a study which is being performed to assess the validity of current LWR piping design practices with specific regard to the specification of loadings. A comparison is made of current design loadings and loadings which have been experienced in operation. The sources of design loadings are currently accepted industry loadings as reflected in LWR Safety Analysis Reports and a limited survey of NSSS/A-E efforts relative to recently recognized loadings. The basis for operational loadings are actual field measured and observed responses for existing LWR piping systems under a full range of operating conditions.

INTRODUCTION

The operating performance of every Light Water Reactor (LWR) plant is heavily dependant upon how well piping systems function. This dependance on piping for trouble-free operation accents the need for comprehensive and accurate design methods. For a typical LWR, a substantial portion of the design process consists of the qualification of piping systems with respect to stresses caused by a variety of external and internal loadings. In view of the magnitude of this effort and in view of the potential consequences of a piping failure in terms of cost (plant availability) and safety aspects, it is imperative that the input to the design process, e.g. load assumptions, be given sufficient attention in terms of feedback from tests and operations to enable a realistic prediction of actual pipe behavior during plant operation. Providing this feedback to the design process would most certainly lead to improvements in LWR operating reliability and safety.

The specific purpose of this presentation is to demonstrate the need for and benefit of a more efficient utilization of LWR operating experience for the detection and evaluation of significant loadings experienced by piping systems. As a basis for the discussion, a list of significant loadings experienced by LWR piping systems is first presented. This is followed by a discussion of specific pipe loading problems which highlight the importance and need of defining more realistic load definitions in the design process. Finally, recommendations for the utilization of LWR operating plant data to improve the piping design process are discussed.

SIGNIFICANT PIPING LOADS

Listed below are piping loads that are considered significant, or potentially significant, based on their magnitude and anticipated frequence of occurrence or the consequences of a failure caused by them.

a. Sustained primary loads, e.g. internal pressure and dead weight.

b. Sustained thermal expansion loads.

c. Thermal transient loads.

d. Seismic loads.

e. Water hammer and steam hammer phenomena [1] [2] [3].

  - Water slug impact
  - Fast valve actuation
  - Flow discharge into initially voided lines
  - Localized flashing of water to steam and subsequent rapid condensation due to pressure fluctuations
  - Check valve closure or delayed opening
  - Water entrainment in steam lines

f. Flow and equipment induced vibrations.

  - Vibrations caused by pulsations from positive displacement pumps

f. Flow and equipment induced vibrations. (cont.)

  - Mechanical vibrations of pumps transmitted to connected piping
  - Flow induced vibrations

g. Piping loads associated with LOCA events.

  - (PWR) Primary loop movement effects on attached piping
  - (BWR) Suppression pool loads: Steam condensation oscillation and chugging, pool swell impact and drag loads

h. Blowdown loads.

  - (PWR) Thrust loads from safety and relief valves from pressurizer and mainstream lines
  - (BWR) Safety/Relief valve discharge through quenchers in suppression pool [8] [9]

i. Jet impingement loads from ruptured high energy pipes.

All attempts at improving the modeling of loads in the design process are aimed at creating more realistic load representations; i.e. the purpose is to reduce the discrepancies between predicted and actual loading conditions, which may consist of both overprediction and underprediction, as well as the omission of significant loads. The most relevant criteria for assessment of the adequacy and accuracy of load assumptions is a direct comparison of predicted load parameters and piping response against actual quantities obtained from plant operation and testing.

When uncertainties exist with respect to load magnitudes a common approach is to assure "conservative" values, i.e. higher load magnitudes or enveloping load cases. The amount of conservatism for individual load cases (e.g. thermal expansion, seismic anchor motions, water hammer, etc.) can thereby be substantial. The "conservatism", however, is not necessarily uniquely defined with validity in both an individual load case and in the overall final design which often comprises a number of different load considerations, with conflicting demands on supporting scheme. A classical example of this conflict is the desired flexible design concept to minimize restrained thermal expansion effects, which contradicts the stiff design concept which is the most common approach to reduce effects due to seismic and other inertia loads. These possibly conflicting needs of supporting systems can actually exist even within one and the same load event, such as a seismic event involving both inertia effect on the pipe and differential relative anchor and support movements. The potential negative effects from enveloping uncertainties by selecting conservative loads are thus twofold:

- "Unnecessary" overdesign with respect to isolated loading, causing added cost and complexity.
- Damaging trade-offs with respect to other load cases having conflicting supporting needs.

Additional load cases, such as those discussed in the following section as having been identified through experienced generic failures, need to be considered in the design process in combination with sustained loads and various occasional loads which could act concurrently, e.g. earthquake loads. This makes it increasingly difficult to accommodate the load effects in terms of pipe stresses within prescribed limits, which further accentuates the need for a systematic feedback process from operational piping performance data back to the original analysis in order to provide a reliable basis for removal of possible unnecessary conservatisms, existing primarily due to the lack of accurate load information.

Certain categories of occasional loads such as seismic excitation and/ abnormal loading under accident conditions do not readily lend themselves to testing and verification with respect to magnitude. To reduce the overall uncertainty in total load and stress magnitudes it is therefore of even greater importance that the sustained and operational loads and those occasional loads that <u>can</u> be subjected to testing, be given special attention with respect to verification of magnitudes used in the analyses.

PIPE LOADING PROBLEMS

To date, the feedback from operational experience has been essentially limited to reports on strongly abnormal piping behavior and actual damages and failures. A number of significant piping loads have been identified through this rather costly type of experience. Typical examples include water hammer and steam hammer phenomena, vibrations caused by flow oscillations and attached equipment, and blowdown load effects. Specific systems and lines, which have been associated with particularly serious generic load effects identified in operation are e.g.:

- Feedwater systems: Severe water hammer effects caused both by water slugs initiated from drained spargers, and by fast valve actuations. Further, cracks in feedwater nozzles to steam generators (PWR) or pressure vessels (BWR).
- PWR charging systems: Cracks and ruptures in discharge lines from positive displacement charging pumps, caused by vibrations.
- BWR Recirculation systems: Cracks and leaks in bypass lines, caused by vibrations and thermal cycling.
- ECCS and RHR systems: Damaged pipes and supports due to water hammer.
- Main steam systems: Damaged pipes and supports due to steam hammer.

These events have traditionally resulted in costly unscheduled outages for repairs and design modifications to prevent reoccurrance. This emphasizes the need for a more comprehensive and systematic utilization of data obtained from testing and operation which would go a long way towards eliminating these types of problems. For new plants, utilization of this data would provide a basis for more realistic load representation at the design phase. For operating plants, it could be used for early detection of unidentified or illdefined loading conditions which could then be dealt with on a scheduled basis to achieve solutions prior to failure. In either case, it is obviously desirable to lessen the reliance on unexpected failures as a detection method for loads.

UTILIZATION OF OPERATING EXPERIENCE

As indicated in the preceding discussions, there is a definite need for providing feedback from LWR operating experience to the definition of pipe loadings. This includes both testing and system monitoring during commercial operation.

Testing of LWR systems includes the various initial test stages, such as Preoperational, Precriticality, Early Criticality, Low Power, and Power Ascention tests. For the purposes of evaluating pipe loading and pipe response, the tests providing the most valuable information are typically the Preoperational and the Power Ascention tests. The acceptability of piping performance is demonstrated during these tests by measurements and observations of e.g.:

- Displacements due to thermal expansion, in discrete temperature steps
- Dynamic response in terms of vibrational amplitudes and frequencies during transients, pump operation, etc.
- Pipe support and restraint performance in terms of reactions and displacements
- Internal pressure and temperature

In general the tests are used primarily to verify the acceptability of the systems, i.e. to demonstrate that critical parameters do not exceed limitations set by the ASME Sc. III Code and other requirements. This means that the feedback, if any, to the piping analysis function most often is limited to those isolated instances involving unacceptable performance such as insufficient clearances, damaged supports, cracked or ruptured pipes, etc.. The full potential of the tests as a means of actually verifying the accuracy of the analytically predicted piping response is thereby not used, since significant discrepancies, both conservative and unconservative, may still exist between predicted and actual piping behavior even though the tests show an "acceptable" behavior. A more detailed, comprehensive evaluation of test results in comparison with analytical results performed on a routine basis, would provide a feedback function which would prove invaluable for the analytical prediction of more realistic load effects.

Observations and measurements made through monitoring of piping system parameters such as temperature and pressure histories, movements, vibrations, etc. during actual operation, constitute the ultimate verification criteria for the loading assumptions and idealizations used in the analysis. This information is an essential complement to testing in identifying and qualifying loading conditions which occur during operation but may not be experienced under controlled test conditions.

In summary, the benefits of utilizing LWR operating experience more efficiently for the evaluation of piping loads are many. These benefits would apply to the specific plants from which information was obtained as well as all plants in the form of generic improvements to the definition of pipe loadings. A listing of the most notable benefits is as follows:

- Elimination of "surprises" at advanced testing stages or in operation.
- Early detection of inaccuracies in analytical load assumptions, allowing for correction in a controlled and scheduled manner, which avoids costly unscheduled outages.
- Provision of reasons for discrepancies between analysis and operation and reasons for observed abnormal piping behavior, to preclude recurrence.
- Provision of a basis for planned repairs, maintenance, and testing to take place during scheduled outages.
- Provision of a basis for overall improved understanding of pipe loading phenomena, which ultimately results in more rational designs and improved safety.

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SESSION XIV

TMI PLENARY SESSION

Chairman

M. H. Fontana - Oak Ridge National Laboratory

TMI - ONE YEAR PERSPECTIVE

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ABSTRACT

The March 28, 1979 accident at the Three Mile Island-2 (TMI-2) nuclear plant was a turning point for the commercial nuclear industry. It is not yet clear whether it was a downward turning point, or an upward turning point, which it has the potential to be for the nuclear industry. One year later the impact of the accident is assessed in four areas of concern to the industry: the nuclear market, both before and after the accident, and indications for the future; industry regulation, as indicated by changes in the NRC; product design changes, both voluntary and required; and industry response, both for ongoing plants and for the TMI-2 recovery.

INTRODUCTION

Webster defines "perspective" as the capacity to view things in their true relationships or relative importance. There is no doubt that with regard to the relative importance of events in the nuclear industry, TMI-2 was crucial. As we now look back at the accident at Three Mile Island and attempt to view it in perspective we must examine several different areas of endeavor. This paper will address four such areas and attempt to identify and analyze the true relation or relative importance of the TMI accident as viewed one year later.

The areas to be addressed are:

- Industry Response to the accident including a description of the activities associated with the recovery and return to service of TMI-2.
- Industry Regulation, from the standpoint of the changes that have resulted from the recommendations made by the Presidential Commission and the Nuclear Regulatory Commission (NRC) Special Inquiry Group, including a discussion of the NRC Action Plan.
- Product Design and Plant Operations, to identify those significant recommendations that resulted from reviews conducted by the NRC, the utilities, the nuclear suppliers, and others.

- The Nuclear Market, from the standpoint of its activity level prior to the accident and the changes that have taken place since March 28, 1979, including a discussion of the pressures on the nuclear market that must be dealt with to achieve a reasonable market level in the future.

INDUSTRY RESPONSE

It is appropriate to discuss the response of the industry to the accident and the currently on-going TMI-2 recovery program. Three separate efforts will be touched on: the Nuclear Safety Analysis Center, the Institute of Nuclear Power Operations, and the TMI-2 Recovery Program.

Last summer, after the accident, the utility Ad Hoc Nuclear Oversight Committee created the Nuclear Safety Analysis Center (NSAC) to be operated by the Electric Power Research Institute (EPRI). NSAC is doing the main industry technical analysis of TMI and the lessons to be learned from it. It is working on an industry-wide program for collecting, evaluating, and disseminating operational experience at nuclear plants.

In September the nuclear industry created the Institute of Nuclear Power Operations (INPO). An office has been opened in Atlanta and they are moving to staff the organization with a planned level of around 200 people by the end of 1980. INPO will set criteria for both operator and management training as well as perform audits of operating utilities. It intends to establish benchmarks of quality for all levels, from top management through the technical people to the operators and technicians. INPO is in the process of establishing criteria or standards and have begun to evaluate individual utilities' operations.

A final point to be discussed is the TMI-2 Recovery Program. Immediately upon learning about the accident on the morning of March 28, B&W mobilized resources to provide support to Metropolitan Edison to stabilize the situation and put the reactor in a controlled cooling mode. B&W established a command center in Lynchburg that was manned around the clock by high-level managers and knowledgeable engineers. This organization was in direct communication with Met Ed, NRC, and B&W personnel located at the site, and provided operating procedures, backup and contingency plans, and other vital input necessary to support on-site operations.

During the period immediately following the accident, other members of the nuclear industry, including utilities, suppliers, and other industry organizations, rallied to support Metropolitan Edison and General Public Utilities. This support took such diverse forms as substitute utility executives to handle day-to-day business which allowed Met Ed and GPU executives to devote full time to the happenings directly involving TMI-2.

Since April 27, 1979, TMI-2 has been in the natural circulation cooling mode, removing heat from the "A" loop steam generator to the condenser. Once the unit was placed in a stable, long-term cooling mode, it was possible to turn attention to the process involved with those tasks necessary to return

TMI-2 to commercial service. This activity, commonly referred to as the Recovery Program, has achieved several notable milestones as of this time.

- In late August the first Reactor Building sump water samples were taken through a hole drilled through the inner flange of an existing containment penetration.
- In late October the EPICOR II system went into operation and began processing contaminated water from the auxiliary building.
- In early November equipment was inserted through a hole drilled in the inner flange of another containment penetration to permit remote video and environmental monitoring of the inside of the Reactor Building.
- In mid-March two Health Physics personnel entered the Reactor Building air lock to conduct various tests including: swipe surveys of air lock surfaces, air particulate samples, radiation readings, and leak tests of the outer door seals. The inner door to the containment was not opened.

In mid-March TMI-2 is still in the stable continuous natural circulation mode. The power level is 205 KW, the primary system average temperature is 150°F and the pressure is 290 psig. With one year completed the remainder of the Recovery Program will involve completion of the following major tasks:

- Containment purge
- Containment entry
- Start containment decontamination - Spring 1981
- Remove reactor vessel head - Spring 1982
- Remove fuel - Spring 1983
- Complete primary system decontamination - Summer 1983

Containment purging of krypton is presently being held up by the NRC, with final decision expected shortly. As the doses involved would be less than NRC standards, the primary concern is the psychological impact on area residents. The containment purging is the major obstacle to beginning the next phase of the cleanup.

INDUSTRY REGULATION

One of the major issues identified by both the Presidential Commission on Three Mile Island and the NRC Special Inquiry Group dealt with the organization of the Nuclear Regulatory Commission and in the role played by the NRC in the overall regulatory process. Questions were raised regarding the extent to which the NRC emphasized its responsibilities related to nuclear safety versus certain other peripheral regulatory responsibilities, such as those related to licensing activities and nuclear exports.

As a result of these studies, recommendations were made regarding fundamental changes in the organization, procedures and attitudes of the NRC.

Both studies recommend abolishing the present five-member commission heading the NRC and replacing it with a single administrator. Both also stress a shift in emphasis to the monitoring of operating reactors to evaluate operating experience and implement required changes. Both recommend establishing a nuclear reactor safety board or commission with oversight responsibilities for both the NRC and the industry. Other recommendations include:

- Transfer certain nuclear related functions such as anti-trust responsibilities and export licensing to other federal agencies.
- Strengthen the licensing project management organization within the NRC.
- Improve the effectiveness, impact, and management of the field inspection program.

With the exception of appointing a new chairman of the commission there are no indications of any substantive changes within the NRC. The mood appears to be one of wait and see President Carter's reorganization plan when it is sent to Congress. Indications are that the plan will retain the present commission but would establish the chairman as the chief executive officer of the NRC with increased authority in appointing positions within the agency. The chairman would also take command of NRC's emergency response team in nuclear emergencies.

In December the NRC presented NUREG-0660, the proposed staff action plan, to the commissioners. The plan, a blueprint for NRC's future, incorporates all the lessons learned from TMI-2 plus the President's Commission and the Special Inquiry Group recommendations. It covers four areas; operational safety, plant siting, emergency preparations and radiation protection, and the NRC organization including the regulatory process. The plan, which is still in draft form, calls for tremendous increases in the area of technical and management support for operating reactors.

An industry group has made an in-depth review of the NRC's Action Plan and has submitted a comprehensive response to the NRC, recommending a priority for the requirements, deferral to a later date of some which had no direct relationship with operating plant safety, and elimination of some requirements where it was felt that implementation would dilute efforts on higher priority items and could in fact have a negative impact on operating plant safety. It is hoped that the NRC will adopt the recommendations by the industry study group. Even so, it is clear that the NRC Action Plan when issued in its final form will impose additional requirements on operating utilities in plant analyses, modifications, training, procedure upgrades, etc. which will involve more than ten thousand man-years for the operating reactors.

PRODUCT DESIGN AND PLANT OPERATIONS

Following the TMI-2 accident, numerous investigations were initiated, including ones by the NRC, a Presidential Commission, NSS suppliers, utilities, Congressional committees, and others. The purposes of these investigations were many, two of which will be discussed here; nuclear plant design from a safety standpoint and utility operations. The main thrust of the investigations has been that, whereas certain hardware related improvements can be made, the primary problem is management related. The President's Commission in their overall conclusion stated:

> "...fundamental changes will be necessary in the organization, procedures, and practices -- and above all - in the attitudes of the Nuclear Regulatory Commission and...the nuclear industry."

The Rogovin Report followed this in their summary with:

> "...the principal deficiencies in commercial reactor safety today are not hardware problems, they are management problems."

The following summarizes, from a general industry standpoint, the more significant changes in product design and plant operations that have been identified.

Plant design studies indicate that changes to the basic configuration of the primary system are not required, however, there are recommended changes in the ancillary systems. These changes generally center in the area of instrumentation and control and include the following:

- Emergency power supply requirements
- Direct indication of valve position
- Instrumentation for detection of inadequate core cooling
- Diverse and more selective provisions for containment isolation
- Automatic initiation of the auxiliary feedwater system
- Auxiliary feedwater flow indication to steam generators
- Increased range of radiation monitoring
- Improved plant iodine instrumentation
- Primary system level indication

Other hardware related recommendations deal with relief and safety valve performance testing, hydrogen recombiners, and plant shielding. Longer range efforts include improving the man-machine interface and reassessing traditional auxiliary system configurations.

In addition to product design changes, additional analytical efforts are required in three specific areas:

- Small break loss-of-coolant accidents
- Inadequate core cooling including low reactor coolant inventory and loss of natural circulation
- Transient and accident analyses including event tree analyses with consequential and operator failures

In the area of plant operations the studies have centered on operator training, plant procedures, and management technical expertise. Recommendations include:

- Provide an on-shift technical advisor to the shift supervisor
- Provide an on-site technical support center close to the control room
- Devote more attention to the writing, reviewing, and monitoring of plant procedures
- Provide an on-site operational support center separate from the control room
- Strengthen the on-site technical capability and management at nuclear plants including the qualification, training, and retraining of operations personnel
- Perform in-depth evaluations of all abnormal events at each plant plus gather, review, and analyze operating experience from other nuclear plants

The recent NRC Action Plan, as mentioned earlier still in draft form, will undoubtedly add to the product design and plant operation changes already discussed. Hopefully it will incorporate the industry study group recommendations in its final issue.

As mentioned earlier, and as can be seen by the above lists of changes, the studies have indicated that the primary problems associated with nuclear power generation are management related and not hardware related. This reinforces the inherent safety of nuclear plants. At TMI-2 the safety systems worked; there was no meltdown and the release of radioactivity was extremely low. The redundancy of equipment, the defense-in-depth concept, paid off and the plant was safely shut down.

THE NUCLEAR MARKET

In 1973 and 1974 fifty nuclear units were ordered by U.S. utilities. As a result of the reductions in growth in electricity demand that followed the oil boycotts and the rapid increases in kilowatt-hour costs, utility purchases of new capacity dropped drastically. From 1975 through 1978 only 12 nuclear units were ordered. Nine of these have already been cancelled and the status of the other three is tenuous at best. In short, the domestic nuclear utility market had shrunk to the point that, at the time of the TMI-2 accident, there were only three domestic utilities considering nuclear steam system (NSS) proposals.

What then was the "true relation or relative importance" of TMI-2 to the domestic nuclear market? That question may best be answered by considering the fact that the three utilities ceased their efforts after the accident and allowed the bids to expire. In all cases there were other factors affecting their decisions but there is no question that the TMI-2 accident was a major one and perhaps the deciding one.

Attempts have been made in Congress and in various states to introduce legislation which would restrict nuclear power growth in one way or another. Notable at the present time are Mr. Udall's bill on licensing and Senator Hart's bill on waste management.

The nuclear industry faces a very serious public acceptance problem at this time. Public concern over the risks of nuclear power has increased and the nuclear issue has emerged in the presidential campaign. Two candidates, Kennedy and Brown, are 100% negative, advocating eventual shutdown of all operating reactors. Regardless of the economic advantage nuclear has over other forms of power generation, until the public acceptance problem can be overcome there will be no new nuclear projects started. The solution is to build public and political trust through the demonstration that nuclear power is safe and that its benefits substantially outweigh the risks. This would create an environment where utilities could be assured that nuclear plants could be purchased, designed, licensed, constructed, and placed into operation on a predictable schedule, at predictable costs, and without undue risks. In order to build trust in nuclear power and create this environment, the industry must, as a minimum:

- Respond appropriately to concerns of the Kemeny, Rogovin, and NRC Lessons Learned reports, demonstrating the managerial and technical expertise necessary to assure safe and dependable plants.

- Mount a comprehensive industry-wide effort to improve the reliability of nuclear plants.

- Reinforce the current program of public information, particularly at the "grass roots" level.

- Work through appropriate channels, both technical and political, to resolve the waste disposal and other major nuclear issues.

SUMMARY

As stated at the beginning, the intent of this paper was to explore the true relation or relative importance of the TMI-2 accident in four of the many varied areas of endeavor within the nuclear industry. One year later this event continues to loom over the industry and impact its ability to regain the appropriate status as a major contributor to this nation's energy future. We, as informed members of this industry, must shoulder the responsibility to see that the lessons learned from TMI are incorporated into the designs, that the public is convinced of our dedication to safety and of our competence in manifesting this dedication, and that from this base we proceed to return nuclear power to its full potential as a contributor to U.S. energy needs.

IMPACT OF THREE MILE ISLAND ON THE NUCLEAR COMMUNITY AND THE FUTURE OF NUCLEAR POWER

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ABSTRACT

The Three Mile Island incident has already impacted on the future of nuclear power specifically and on the energy crisis generally. The resultant NRC's de facto moratorium on the licensing of commercial nuclear power plants will increase the use of oil, raise the cost of electric power, and may create power shortages. On the positive side, TMI may have been beneficial in that it has precipitated a searching reassessment and improvement of nuclear power safety and practices-TMI not withstanding, the rate of nuclear power expansion must increase; otherwise the energy crisis will deepen in the years to come.

Introduction

Not suprisingly, the Three Mile Island incident of March 28, 1979, whose fallout seemed to not only include everything from proliferating anti-nuclear demonstrations to rising box office receipts for "The China Syndrome", has also had an adverse impact on the nuclear community. It will undoubtedly impact on the near future of nuclear power specifically, and all energy developments generally.

Nuclear Industry Prior to TMI Incident

Before TMI, some 206 plants, with a capacity of about 200 gigawatts, represented our Nation's commitment to nuclear power.

A look at the trend of nuclear generating equipment contract awards by utilities prior to TMI, shows that orders were flat from 1975 to 1977 and then declined in 1978, to no orders in 1979. The decline in nuclear orders was attributed to a lack of perceived Administration support for nuclear power, uncertainties in the plant approval process, and the fact that capital investments sometimes approached the net worth of utilities, thus making the risk too great.

Today's operating nuclear plants were purchased in the 60's, and the decision then to go nuclear was based on the engineering judgment that such plants with lower fuel costs would be more economical than competing fossil units. Generation costs proved to be more economical, just as predicted. The AIF released actual generation costs from utilities showing nuclear costs steady at 1.5¢/kWh, coal rising to 2.3¢/kWh and oil at greater than 4.0¢/kWh for 1978. Estimated costs for 1979 based upon increased fuel costs and changed capacity factors show nuclear at 1.8¢/kWh, coal at 2.6¢/kWh and oil at 5.2¢/kWh.

As for fuel costs, fossil fuel prices have risen sharply since the 1973 oil embargo. In the past ten years, crude oil has risen 10-fold, natural gas 8-fold and coal increased 4-fold. Uranium prices have also risen, but their effect will be felt only gradually because of the nuclear fuel cycle characteristics and the smaller effect of the fuel component in the total nuclear generation cost picture.

The commercial capability of the U.S. nuclear industry is somewhere around 30 sets of nuclear plant hardware and engineering per year. Given a reasonable mix of domestic and export business, most agree that the engineering, equipment, construction and other necessary capacities exist to support this 30-set profile for the future. But TMI could change the profile for the future.

In the past, the public favored the construction of nuclear power plants. In fact, the trend for those "favoring" and "opposing" was approximately the same until the last quarter of 1978, with those "opposing" increasing in numbers. Nuclear power advocates still exceed those who are against, but by a smaller margin.

TMI and the Energy Crisis

The immediate effect of TMI on the energy crisis was the cancellation of about 20 nuclear plant orders. Nuclear plants now under construction are likely to encounter delays and no significant increase in operating licenses will be issued until it is known what changes are to be made.

In the much publicized waste issue, a study of nuclear vs. coal wastes, equated to the same plant size, nuclear comes up with a far better showing. Nuclear wastes are five million times smaller by weight and a billion times smaller by volume. The commensurate health effects of nuclear on the public is, therefore, miniscule. The important thing to note is that in reality, waste disposal is needed, independent of the continued use of nuclear power.

More significant was that the incident has intensified the national debate on nuclear power. The Administration is moving slower on such issues as waste disposal, away-from-reactor fuel storage and the timing for commercial fast breeder reactors. The Price Anderson Act will come under closer scrutiny, and licensing reform will probably slow down to a crawl.

Prior to TMI there was some evidence that the licensing process was beginning to stabilize. That TMI will impact on the licensing process of future plants is a foregone conclusion. Nuclear licensing reform is still needed to speed up the nuclear plant construction process and in the long run would help alleviate the energy crisis.

Confusion, fed by conflicting reports from official sources, and frequent inaccurate media coverage focused public attention on radiation doses as a result of the incident. While government findings revealed that none of the residents really received much more radiation than Colorado residents receive from natural sources, the fear of radiation is becoming an increasing concern to the public.

The impact of TMI on the energy crisis is that in the near term it will probably slow down the expansion of nuclear power. In the long run, however, TMI will provide beneficial effects --- in that a great deal will be learned from the incident and that it will help Americans be more aware of our limited options and to validate nuclear power as a vital energy source.

Energy Crisis - The Future

For new energy supplies, the Nation has now moved into "real time". Supply issues are now no longer academic and need to be faced up to since energy shortages are approaching serious proportions. High interest rates and inflation will slow down investments in conservation measures. Uncertainties of Middle East oil supplies are likely because the continued political foment in the area will decrease the availability of gasoline and heating oil. The EEI recently stated that the slow down in nuclear power will create power shortages in the early 1980's. These increased shortages in the next decade will undoubtedlv result in deep social problems.

Projections show that 10 years hence, our dependence on foreign oil could increase to 13-14 million bbl/day, at a cost of $150 billion per year by the late 1980's, if nuclear power and synthetic fuels are not accelerated.

Gasoline supplies could be increased if utilities are permitted to use coal and uranium fuel rather than oil for electrical generation. If all electric power was replaced by nuclear and coal, then the Nation could expect a 20% increase in gasoline supply.

An optimistic note in the energy crisis was a pledge by industrial nations attending the Tokyo Summit Conference in June 1979 to place ceilings on oil imports, to increase coal use, to expand nuclear power generating capacity, and more important, to develop new technologies using large public and private resources.

While 1979 was a depressing year for nuclear power, the winds of change in 1980 indicate that the prospects support continued growth of safe nuclear energy. During a recent White House meeting, scientists were informed by Administration officials that an additional 90 nuclear plants have to be ordered and operating by the 1993-95 period. This new capacity would mean that over 300 GWe of nuclear power will be on the line by the year 2000. The Congress is also moving to pass bills that will accelerate the demonstration of high level nuclear wastes.

The operator's response to contain radioactivity at the Crystal River Plant was encouraging, in that the lessons learned from TMI are being followed to insure public and plant safety. The NRC is also beginning to open up the licensing logjam by their recent approval for TVA to start up their Sequoia Unit.

The INFCE study was initiated by President Carter because of his concern for proliferation. All the nuclear power countries were involved in the study and they recently concluded that the proliferation risks of reprocessing was not any greater than enrichment and that there was no alternative reactor or fuel design that reduced the proliferation risks. All the countries, with the exception of the U.S., supported the continued development and early commercialization of the fast breeder reactor.

Finally, the CONAES study of the National Research Council concluded that coal and nuclear power were the only economic alternatives for large scale electric power development for the remainder of the century. The study also concluded that there should be continued development of the LMFBR for early deployment by the next century. As far as proliferation, CONAES, like INFCE, stated that there was no technical fix, including the halting of nuclear power to avert diversion of materials for nuclear weapons. Coal was also considered to have a greater risk to the public than nuclear power.

If nuclear power is not expanded, my forecast for future electricity demand and generation for the year 2000 shows a shortfall in needed generating capacity. Such a program would result in an increase in the use of oil for electrical generation, rather than phasing out such use as desired.

Synthetic fuels and expanded use of coal for electric power do not appear to be a panacea to the Nation's future energy problems. Engineering problems, astronomical costs, major environmental issues, and doubts about their energy value will require careful analysis.

Tripling the use of coal by the year 2000 is fraught with problems, too. Of particular concern to the scientific community is the possible "greenhouse effect" due to global carbon dioxide buildup and its resultant effects on regional climatic changes.

Conclusion

Electricity demand is growing twice as fast as total energy demand. Both are still inexorably tied to economic growth. Therefore, if our Nation is to grow and prosper, we must speed the use of technologies we presently understand and accelerate new systems to commercialization, TMI not withstanding. Otherwise, the energy crisis of today will deepen and will continue to be with us for some time to come.

FINDINGS AND RECOMMENDATIONS OF THE NUCLEAR REGULATORY COMMISSION'S SPECIAL INQUIRY GROUP ON THREE MILE ISLAND

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ABSTRACT

The Special Inquiry Group found that the greatest area of risk in operational reactor safety today in the United States is the human element, and that reducing that risk has been underemphasized compared to improving design and hardware. The Group recommended a new philosophy of operator training; strengthening the on-site technical capability and management of operating companies; new requirements for qualified engineer supervisors on every shift in the on-site supervisory management chain; chartering of an operating consortium with the capability to operate plants of a number of utilities on either a contract or "receivership" basis; and a revised and upgraded safety inspection system. The Group also recommended new future policies for remote siting, emergency planning and design review, including increased use of quantitative risk assessment.

INTRODUCTION

The findings and recommendations of the Nuclear Regulatory Commission's Special Inquiry Group can be roughly divided into two categories. The first are those aimed at upgrading regulatory standards -- or developing new ones -- to improve the safety of operating reactors, chiefly through improving the qualifications, the training and the overall competence of site operations personnel. The second group of recommendations looks to changes in the regulatory agency itself, in order to ensure that it has the structure, the management and the capabilities necessary to develop and apply these kinds of requirements effectively. Although the Group's recommendations relating to NRC reorganization have received a great deal of attention from the public, they are really means to an end -- improving operational safety. Therefore I would like to focus on the substantive changes the Special Inquiry Group believed were necessary to improve safety directly.

A NEW PHILOSOPHY OF TRAINING

It is noteworthy that both of the "independent" investigations into the Three Mile Island accident -- that of the Kemeny Commission and the NRC's own Special Inquiry -- reached similar conclusions: namely, that the weak element, the greatest area of risk, in operational reactor safety today is the human element, and that reducing the risk in that area has been severely underemphasized compared to improving the hardware element, the design and manufacture of engineering systems. Considering how sophisticated and potentially dangerous a technology is involved, the investment that has been made over the years by the commercial nuclear industry in the human control element -- investment in terms of both attention and dollars -- is relatively very small compared to the massive investment in design and equipment safeguards. By human control element, I mean not only investment in operator training and the salaries paid to operators, supervisors and managers of nuclear operations, but also investment in sophisticated information display systems, in management control, and in continuing education of site operations personnel about operating problems at other plants.

For example, the Special Inquiry Group found that the fact that a Reactor Operator or Senior Operator has passed the tests to receive an NRC license by no means ensures that he is competent to control a commercial reactor safely when something unexpected occurs. Only effective training can guarantee this. Yet, operator training has generally been a "backwater" both at the NRC and in the industry as a whole.

The weakest aspects of operator training clearly are (1) the emphasis on training for <u>normal</u> operations, rather than transient or accident situations; (2) the emphasis on classroom and book training as opposed to simulator or "hot" (operational) training; and (3) the lack of training and operator qualifications in analyzing the way the reactor system as a whole functions -- or might misfunction.

The Special Inquiry Group contrasted the control room operator at a commercial nuclear power plant to an airline pilot. The vast majority of the pilot's time is spent in routine, high-altitude flying. Similarly, the reactor operator's typical 8-hour shift is a study in boredom: in normal operation the plant virtually runs itself. What does require considerable pilot skill is takeoffs and landings. Similarly, in a nuclear plant, routine startup and shutdown are complex procedures in which a variety of coordinated actions must be taken and instruments closely monitored. Reactor operators, like pilots, are extensively trained for these manipulations.

However, the public expects a commercial airline pilot to be trained and qualified not only for routine operations, and for takeoffs and landings, but for emergencies and accidents as well -- the loss of an engine, sudden depressurization, hydraulic failure, or an engine fire. It is in the analogous area that reactor operator training has been seriously deficient. Other than being required to memorize a few emergency procedures, reactor operators have not been extensively trained to diagnose and cope with a variety of equipment malfunctions, serious transients and accidents.

The Special Inquiry Group Report describes one simulator training session developed after the accident by Metropolitan Edison and Babcock & Wilcox personnel that is the type NRC regulations should be designed to require. In that session, the simulator was programmed not to deal with just startups or "routine" transients but with various accident sequences. Then, instead of a crew of Met Ed operators splitting up so that some studied books while others worked on the simulator, the entire crew was taken through a number of accident sequences and graded on their responses as a team, not as individuals.

Some of these accident sequences were not limited to a single failure, but were multiple failure accidents. They were not, as is customary, "short" accidents. The simulator was programmed to play accidents out over a long period of time. And the crew was not told beforehand what casualities would be programmed. The goal was not to "beat the game" but to limit the damage -- failure after failure was "sent in" by the programmers to see how well the operators could diagnose and react to them.

ON-SITE ENGINEER SUPERVISORS

Improved operator training alone will not be enough to guarantee that site operations personnel possess the requisite technical capability to handle emergency situations. Under current NRC manning requirements, only one Senior Reactor Operator and a handful of other operators are required to be on duty at any given time; often, especially during night shifts, a shift supervisor may be the senior person on site. Many foremen and shift supervisors in today's plants have worked their way up from being reactor operators; they may not have the depth of engineering and technical background to be able to analyze the reactor system and prescribe the correct actions in the event of an emergency.

In the nuclear Navy, enlisted men serve as reactor operators, but they are supervised on the spot by engineer-officers 24 hours a day. Operators of the larger and vastly more complicated commercial plants and their supervisors are the equivalent of these enlisted men. The extra training and experience they have acquired in the commercial program often cannot substitute for an engineering background. And training alone cannot make control room operators into the equivalent of the Navy's "engineer officers of the watch."

Thus, the Special Inquiry Group recommended, in addition to substantially increased manning requirements, the development by NRC of <u>new</u> regulatory criteria requiring that capable engineers with knowledge of the plant equivalent to that required of licensed operators be placed in charge of the operating crew on every shift. These engineers would be certified or licensed by the NRC both for their engineering qualifications and for their intimate knowledge of the particular plant's characteristics.

IMPROVED INFORMATION DISPLAY

Finally, it is essential that the competent site operations team have available to it quickly and reliably information about plant conditions necessary to cope with a transient or developing accident. The Special Inquiry Group found that the major problem in this area was not so much that the instruments did not exist, but that the information theoretically available could not be obtained and displayed in the control room in timely, usable fashion to respond to a casualty.

The Group's Report therefore recommended that every plant be required to install the equivalent of an on-line reactimeter that could be constantly monitored both in the control room and, through telemetry, in off-site locations.

NRC MANAGEMENT

What is required on the part of the Nuclear Regulatory Commission to implement these changes is not an incremental increase in existing regulatory requirements, but some new approaches. For example, the Special Inquiry Group suggested that substantially improved training probably cannot be guaranteed unless the NRC takes a more direct and substantive role in the training of operators, much as the FAA does in the training of pilots and flight controllers. This might include the certification of training facilities; establishment of a minimum curriculum; substantive review of the operator procedures that are drilled during training; and certification of instructors. All of this amounts, in essence, to a new philosophy of regulation in the area of training.

Similarly, certification of on-site supervisors and managers would require an entirely new program within NRC: new regulatory criteria of the kind NRC has never before developed, and new talents within the agency to formulate and apply these criteria.

To take another example, the Special Inquiry Group called for a new system to evaluate operational experience. This will require the integration of changes in several areas: a new office within the agency, with new authority; and a new reporting system with new follow-up requirements, to separate important events from trivial ones and obtain more in-depth information about the former.

These new approaches probably cannot be developed by the NRC staff without strong direction from the top -- that is, command decisions by the Commissioners themselves. That is why the Special Inquiry Group stressed the importance of strong central management within the agency.

Better management is also required to upgrade and improve the current safety inspection program. The Special Inquiry Group made a number of recommendations in this area. They included greater use of the "team" or "blitz" inspection technique; more emphasis on periodic overall evaluation of the safety of each plant; and increased monitoring of the on-site management and technical capabilities of utilities. The latter two activities would also require development by the NRC of new expertise within the agency itself, to be able to assess and evaluate the quality and competence of the onsite operating crews and their engineering supervisors.

A NATIONAL OPERATING COMPANY OR CONSORTIUM

In view of the weaknesses in operational safety found by the Special Inquiry Group, we suggested that consideration be given to the chartering of a national operating company or consortium. The company or consortium would itself obtain a license from the NRC and would be available to contract with individual utility licensees to operate their nuclear plants on an ongoing basis. Establishment of such an enterprise would also afford the NRC the opportunity to *require* a utility that did not have the capital or expertise to meet upgraded safety requirements to place operation of its plant in the hands of the consortium, or to seek the consortium's assistance as a condition of continued operation.

REMOTE SITING

The Three Mile Island accident demonstrated that the evacuation of people living within a 10-mile radius of a commercial nuclear powerplant, or beyond, needs to be considered a realistic precautionary measure, even when observed levels of radioactive release are well below previously formulated Federal "protective action guidelines." For some years the NRC has been moving informally toward requiring new reactors to be sited further away from large population clusters. However, the formal siting requirements used by the agency do not adequately reflect this concern: they provide that reactors be sited within a "low population zone," a very small area dependent upon the design features of a plant, with a radius of only a few miles or less around the plant. The low population zone at Three Mile Island was an area within 2 miles of the reactor itself. During the accident both State authorities and the NRC talked of the need for evacuations encompassing areas 5, 10, and possibly even 20 miles from the Island.

In the past, the NRC has consistently regarded "engineered safeguards," i.e., automatic emergency safety systems within the plant, as a permissible tradeoff permitting the location of a plant near a heavily populated area. That is, the plant's safety equipment, combined with the containment structure and the ability to evacuate the low population zone, was deemed sufficient to protect public health and safety. Our analysis of how close the accident at Three Mile Island came to a situation in which evacuation might have been required on a precautionary basis, at least, led the Special Inquiry Group to conclude that this philosophy simply is not valid. Evacuation must be considered as an independent means of protection for citizens living near a nuclear plant, over and above the engineered safety systems designed to mitigate an accident and to prevent releases. In the case of siting, "distance" should be regarded as the ultimate defense-in-depth barrier protecting those who live near nuclear plants.

For future reactors, the Special Inquiry Group concluded that siting should be restricted to areas at least 10 miles from any significant center of population. With respect to existing plants and those under construction, however, changes would have to be made in evacuation planning within such an area.

EVACUATION PLANNING

Until March 28, 1979, planning for evacuation around nuclear plants by Federal, State, and local authorities was uneven, at best. The NRC itself did little to encourage such planning, in large part because of a prevailing attitude that a serious accident with releases beyond containment simply would not happen. The NRC has required utility company licensees to plan only for protective measures within the low population zone, and to show that they have their own emergency plans which include notification to and coordination with local and State authorities. The existence of an effective State emergency or evacuation plan in case of accident has not been a condition for granting a reactor operating license. Under current regulations, States may submit plans to the NRC for approval, but at the time of the accident only a few States had NRC-approved plans in place, and Pennsylvania was not one of them.

Because the Special Inquiry Group believed that protective action -- ranging from staying indoors to partial evacuation to general evacuation -- must be considered an independent form of safeguard, it recommended that workable evacuation plans be made a prerequisite to continued operation of existing and future reactors. The success of such an approach, however, obviously depends upon what is regarded as an "adequate" plan. The Group recommended that the emergency plan should not be just an abstract document. Rather, as a condition of the operating license, it should be viewed in the same fashion as an engineered safety system in the plant. The typical plant's technical specifications provide that when engineered safety systems become "degraded" or inoperable, the plant may have to be shut down if the situation cannot be remedied within a short period of time. Whether an evacuation plan can realistically be executed at a particular time should be treated in the same fashion. Thus, if a 5-foot blizzard makes roads in the area of the plant impassable, the utility should be required to notify the NRC immediately. The NRC would then, after consultation with other Federal and State authorities, make the decision whether the plant should be shut down (or some other measure instituted, such as a decrease in power level) until the evacuation plan once again became workable.

IMPROVED BASES FOR DESIGN REVIEW

Finally, the Special Inquiry Group found that while the current Design Basis Accident approach to evaluating the safety of reactor designs had worked well in dealing with a new technology, enough experience has now been gained to supplement that approach very substantially with quantitative risk assessment techniques.

CONCLUSION

The Special Inquiry Group recommended many changes in the existing reactor safety program, the most urgent of which focus upon the training, qualifications and competence of on-site operations crews and their management. Whether these changes will be made depends primarily upon the seriousness of the regulatory agency in implementing new requirements and upon the dedication of industry to upgrading the quality of operations.

SESSION XV

TMI-2 : a) RISK ASSESSMENT AND IMPACT
b) THERMAL HYDRAULICS

Chairmen

L. Pease - Atomic Energy of Canada, Ltd.

J. F. Quirk - General Electric Company

Dup

EVALUATION OF THE THREE MILE ISLAND ACCIDENT IN THE CONTEXT OF WASH-1400

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ABSTRACT

A comparison of the WASH-1400 Reactor Safety Study with commercial reactor experience shows that the Three Mile Island accident does not challenge the validity of WASH-1400. The severity of the accident was consistent with a "PWR-8" category accident as described in the study. The exact sequence of failures in the TMI accident is not included in WASH-1400, because of design differences between the reference Westinghouse PWR used for the study and the Babcock and Wilcox PWR at TMI. However, similar sequences are included in WASH-1400 and the TMI sequence is included in the WASH-1400 general description of transient-initiated accidents. A probability analysis shows that the occurrence of the TMI accident after about 400 reactor years is consistent with WASH-1400 probability estimates. The impact of the accident on cancer statistics is estimated and compared to the expected public perception of health effects.

INTRODUCTION

The accident at unit 2 of the Three Mile Island nuclear station (TMI-2) on March 28, 1979, occurred after approximately 400 reactor years (RY) of commercial nuclear reactor operation in the US. The purpose of work summarized here was to evaluate the probability statements in the WASH-1400 reactor safety study (RSS)[1] in view of the TMI-2 event and to estimate the likely public impact of TMI-2. The RSS probability estimate for such a release was found to be consistent with the fact that the TMI-2 accident occurred. The expected health effects are consistent with those for a low-level category of radioactivity release as described in the RSS and they are immeasurably small. However, the public perception of the health effects of the release is likely to be much more severe than the estimated health effects.

CATEGORIZATION OF THE TMI ACCIDENT

The nature and severity of the accident coincide with a category of radioactivity release for a pressurized water reactor (PWR) described in RSS as "PWR-8." A PWR-8 release may involve damage to the nuclear core but without substantial fuel melting. Radioactive fission products residing in intragranular fuel gaps could escape into the primary-system coolant, and coolant could escape the primary system through a small breach. (The primary system includes the nuclear core, the cooling water flowing through the core, and the vessel and piping that contain the water and the core.) Failure of the containment to isolate properly (i.e., prevent the escape of large amounts of radioactive material) then leads to a release to the environment. The severity of a PWR-8 release is not regarded as significant enough to require evacuation in the vicinity of the reactor. (The lowest release category is PWR-9, which involves proper isolation. The higher categories, PWR 1-through PWR-7, involve core meltdown).

The primary releases from TMI-2 occurred because of operation of the make-up and let-down flow of primary coolant and improper operation of waste gas venting in the days following the accident.[2] Operation of this system made it necessary to violate containment in a manner that caused radioactive xenon gas to escape the containment due to entrainment in the primary coolant. The subsequent noble gas release to the environment of 2.4-13 million curies was equivalent to 1-4 percent of the core inventory of 302 million curies. Releases of radioactive iodine were 13-17 curies, or about 0.00001 percent of the core inventory of 152 million curies, which is equal to the estimate for iodine release stated in WASH-1400 for a PWR-9 category release. However, the noble gas release is 4 to 20 times higher than the WASH-1400 estimate for a PWR-8 release (0.2 percent). Thus TMI-2 did not involve proper containment isolation. The radio-iodine release was consistent with category PWR-9. Severe fuel damage and containment violation, however, caused a significant noble gas (radio-xenon) escape consistent with core melt categories. The combined effects of the low iodine and high noble gas releases are consistent with category PWR-8.

THE TMI ACCIDENT SEQUENCE

The sequence of events that caused the TMI-2 accident are, stated simply; (1) turbine trip, (2) normal opening of the PORV, (3) stuck open PORV, and (4) throttling of safety injection. This sequence does not appear in WASH-1400, but a similar sequence does which involves failure of auxiliary feedwater causing the PORV to open. The distinction is that in the reference Westinghouse reactor of WASH-1400, the PORV does not normally open following turbine trip, as it did in the Babcock and Wilcox reactor at TMI, because anticipatory reactor trip in the Westinghouse design mitigates rapid pressure rise. Thus the additional failure of auxiliary feedwater is included in the WASH-1400 sequence as a prerequisite to PORV opening. (Note that the immediate unavailability of auxiliary feedwater at TMI did not contribute to the accident other than to cause confusion, because the PORV would have opened anyway. The failure would have been significant if the PORV were not normally required to open.)

The general description of transient-initiated accidents in WASH-1400, however, does encompass the TMI-2 accident. The "Functional Event Tree - PWR Transient Events" shows the result of a transient with insufficient heat transfer to the environment during cooldown to be "eventual core melt, if no operator action taken." At TMI-2, operator action to open the pressurizer PORV line block valve at 192 minutes after turbine trip and to reinitiate safety injection at 200 minutes prevented core melt. Thus accidents similar to the TMI-2 accident were not overlooked in WASH-1400. In fact, those transient-initiated accidents that are postulated to result in core melt are large contributors to reactor accident risk as calculated in WASH-1400.

THE PROBABILITY OF AN ACCIDENT

To compare the occurrence after approximately 400 RY to the predictions of the RSS, a series of questions must be considered. These involve various levels of detail in describing the accident. First, what was the probability of an accident in either a PWR or BWR (boiling water reactor)? Then, what was the chance that an accident would have occurred in a PWR instead of a BWR? Finally, what was the chance that the PWR accident would have resulted in the level of damage and public radiation exposure realized at TMI-2 instead of more severe consequences arising from core melt? These questions are addressed in the following paragraphs.

The RSS's best estimate of the probability of an accident involving reactor core damage and radioactivity release is one in 2 000 yr per reactor for PWRs and one in 7 750 yr per reactor for BWRs. This suggests a 13% chance (approximately 1 part in 8) of having realized at least one such release in the US by March 28, 1979, after approximately 223 RY in PWRs and 187 RY in BWRs. Probability estimates normally are not rejected in statistical analyses until the probability of the observed event(s) would be lower than 1 part in 20 (i.e., "95% confidence level" in hypothesis testing). Thus, the fact that an accident involving core damage and radioactivity release occurred is consistent with RSS probability estimates.

The probability that such an accident would have occurred in a PWR instead of a BWR is about 4 parts in 5. This is due to the higher estimated probability of accidents in PWRs by nearly a factor of 4, and the 20% more reactor years of operation in PWRs. Thus, the occurrence of an accident in a PWR was more likely.

The nine categories of radioactivity release for PWR accidents have different relative probabilities of occurrence, with the less severe releases having the higher probabilities. The probability of the PWR-8 category release instead of any other is 8%, or about 1 part in 12. This is the second most likely category. (The most likely outcome, a PWR-9 release, has a probability of 80%.) The occurrence of a PWR-8 category release is consistent with the RSS probabilities.

While the occurrence of TMI-2 is consistent with RSS probabilities, it should be noted that the data can be used to support other probability statements as well. For example, the fact that an accident has occurred, releasing radioactivity and damaging the nuclear core after 400 RY, indicates that the probability of an accident could be as high as 1 in 130 per reactor-year (as compared to the RSS estimates of 1 in 2 000 for PWRs and 1 in 7750 for BWRs). If the probability were higher than this,it would have been unlikely (i.e., less than a 5% chance) to have had only one accident in 400 RY. However, if the probability of an accident were lower than 1 in 7 800 yr per reactor, it would have been unlikely to have had as many as one accident in 400 RY. Thus, the data support (with 90% statistical confidence) accident probabilities in the range 1 in 130 to 1 in 7 800 per RY. Because existing data on reactor accidents are limited to one event, uncertainty is wide in probability statements made on the basis of the data. In these examples no generic distinction has been made between BWR and PWR probabilities, because the limited data indicate that the probabilities are not necessarily different. The best-estimate probabilities and the conservative probabilities (i.e., 10 times higher than the best estimate) from the RSS lie in this range.

HEALTH EFFECTS OF TMI RELEASE

The WASH-1400 best estimate of population exposure to radiation is 920 rem for PWR-8 releases. This number is related to expected increases in the incidence of cancer and genetic effects in the population, because the exposure is well below acute illness or fatality limits. The maximum value for a PWR-8 release is listed in the RSS as 15 000 rem (total population dose). The cancer rate from low-level radiation is estimated to be 100 incidents per 1 000 000-rem population exposure. (The rate for genetic effects is conservatively assumed in the RSS to be the same as that for cancer.) This assumes that risk of radiation-induced cancer or genetic effects is directly proportional to radiation dose. Hence the RSS average cancer incidents range from the best estimate of 0.09 to the maximum 1.5. This is equivalent to a probability ranging from a best estimate of 10% to a maximum of 80% of one or more incidents of cancer. The same results apply to genetic effects.

First indications were that the TMI-2 release resulted in an exposure of 3500 person-rems. The estimates of the number of resultant cancer incidents as stated on page 31 of the report of the President's Commission on the Accident at Three Mile Island-2 are based on this number. Revised estimates of 2000 person-rems were finally made by the Commission's staff. The recent National Academy of Sciences (NAS) estimate for cancer rate is one per 5 000 rem,[3] which could be as much as fivefold high or low, and which is twice as high as the RSS estimate. Given the NAS rate, 7 000 of the 360 000 annual cancer deaths in the US are attributable to background, low-level radiation sources (for example, atmospheric radon, cosmic rays, medical uses of radiation, natural radio-potassium in the human body). Thus, the average number of cancer incidents from TMI-2 is roughly 2 000/5 000, or 0.4. This is equivalent to a 67% chance of zero incidents, a 27% chance of one, a 5% chance of two, and a 1% chance of three or more. These results are based on a Poisson distribution with an average of 0.4.

The societal impact can be evaluated by estimating the effect on life expectancy in the US from low-level radiation exposure from reactor accidents. Based on the estimated health affects of TMI-2, the life expectancy in the US would be decreased less than 2 h if one TMI-2-type accident occurred every week. Further, the rate of genetic effects would increase less than 0.1%.

The perceived impact on the public may be greater than the estimated health effects warrant because a single incident of a radiation-induced health effect is usually not attributable to a specific source (TMI-2, cosmic radiation, atmospheric radon, etc.). The normal pre-accident cancer death rate among the approximately 2 000 000 persons living in the vicinity of TMI-2 is about eight per day. Because many cancers are curable, many more than eight people per day discover they have cancer. It is likely that the public will attribute many of these to the reactor accident. The normal rate of genetic effects (deformities and genetic-related diseases) for the same population is 100 000 per generation.[1] Many of these may be attributed to the accident, although the increased rate of occurrence of genetic effects from the accident is about the same as that of cancer.

CONCLUSIONS

The fact that WASH-1400 includes accident probability estimates consistent with the occurrence of the TMI-2 accident and the fact that transient-initiated accidents were described in WASH-1400 and found to be major risk-contributors suggest that the TMI-2 accident should not have been a surprise to the nuclear community. While it was not expressly emphasized in WASH-1400, the warning was published four years before the accident.

ACKNOWLEDGMENTS

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THE ACCIDENT AT THE THREE MILE ISLAND UNIT 2 FACILITY AND THE ENSUING ACTIONS BY THE NRC

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ABSTRACT

The accident at Three Mile Island Unit 2 on March 28, 1979, is clearly a major milestone event for nuclear power with worldwide impact. The full impact of the accident technically and otherwise is now beginning to emerge as the findings of various investigative groups (by industry and the government) become available. A Lessons Learned Task Force was established in the Office of Nuclear Reactor Regulation to make early recommendations regarding actions to be taken following the accident. The actions of the Task Force are discussed in the context of short- and long-term phases which involve new and specific requirements for nuclear power plants and to consider the more fundamental issues of nuclear reactor safety based upon the experiences gained from the accident. In addition to the activities of the Task Force, additional actions are being considered in an overall integrated Action Plan now under development by the NRC. The plan will conform significantly to the Presidential Commission's recommendations as well as those of the ACRS and the NRC's Special Inquiry Investigation following reviews by the ACRS and the Commission.

GENERAL

On March 28, 1979, the Three Mile Island Unit 2 (TMI-2) nuclear power plant experienced a loss of feedwater transient that led, through a series of events, to a partially mitigated loss-of-coolant accident with significant core damage. The sequence of events involved equipment malfunctions, design deficiencies and human errors, each contributing in varying degrees to the ultimate consequences of the accident.

Over the past year since the accident at the TMI-2 facility, the NRC staff has been conducting an intensive review of the design and operational aspects of nuclear power plants and the emergency procedures for coping with potential accidents. The purpose of these efforts was to take certain actions in the short-term that would reduce the likelihood of the recurrence of a TMI-2 accident as well as to improve the overall level of safety in nuclear power plants. It is clear that major actions are necessary to ensure a low likelihood of a repeat of the TMI-2 accident. Some of these actions were in use at the time of the February 26, 1980 incident at Crystal River Unit 3 which lends support to their effectiveness.

There are a number of other investigations concerning the TMI-2 accident. As a result of these efforts, a number of reports [1] [2] [3] [4] have been published by the NRC that deal with certain safety aspects of the accident and bear on the broad question of safe nuclear power. The Presidential Commission issued its report in late October 1979 [5]. The NRC's Special Inquiry Group issued its report in January 1980 [6]. In addition, several Congressional inquiries are in progress and the industry is evaluating major aspects of the accident. Generic reports have recently been issued by the staff that deal with the results of the Bulletins and Orders Task Force generic reviews of feedwater transients, small break LOCAs and other TMI-2 types of events [7] [8] [9] [10].

The NRC realized that it was not necessary to await the outcome of these investigative groups to identify some of the significant lessons resulting from TMI-2. Consequently, in May 1979, a TMI-2 Lessons Learned Task Force was established. It was an inter-disciplinary team consisting of 22 professionals from the Office of Nuclear Reactor Regulation, Nuclear Regulatory Research, Inspection and Enforcement, and Standards Development. Its purpose focused on the identification and evaluation of those safety concerns originating from the TMI-2 accident that require licensing actions. The work of the Task Force was essentially completed in October 1979.

In general, the TMI-2 Lessons Learned Task Force focused on identifying actions which go beyond those clearly specified in IE Bulletins and (Commission) Orders [directed toward the operating B&W plants] and which would be applicable not only to operating plants but also to pending operating license (OL) and construction permit (CP) applications.

The Task Force was charged to review and evaluate investigative information, staff evaluations of responses to IE Bulletins and Orders, Commissioners' recommendations, ACRS recommendations, staff recommendations from NUREG-0560 [1], and recommendations from outside of the NRC. In addition, the Task Force was charged to identify, analyze and recommend changes to licensing requirements and the licensing process for nuclear power plants based on the lessons learned. The scope of the Task Force included the following general technical areas:

- Reactor operations, including control rooms, operator training and licensing;
- Reactor transient and accident analysis;
- Licensing requirements for safety and process equipment, instrumentation, and controls;
- Onsite emergency preparations and procedures;
- NRR accident response role, capability and management; and
- Feedback, evaluation, and utilization of reactor operating experience.

The Task Force set its work into two distinct phases; a short-term and long-term plan. The first phase dealt with the development of recommendations for short-term actions which when combined with other requirements, e.g., the IE Bulletins on TMI-2, would establish short-term requirements to ensure the safety of plants already licensed to operate and those to be licensed for operation in the near future.

The second phase considered broader and more fundamental questions in the design and operation of nuclear power plants and in the licensing process. The issues considered are grouped in four general categories: general safety criteria, system design requirements, nuclear power plant operations and nuclear power plant licensing. Recommendations for near-term changes in off-site emergency preparedness and other licensing are under development by others.

SHORT-TERM RECOMMENDATIONS

The Task Force in determining which safety issues required short-term licensing action versus those that could be deferred for further evaluation by the Task Force or others considered engineering evaluation and qualitative professional judgment of the safety significance of the various issues. In this regard, the Task Force selected items for "short-term action" if their implementation would provide substantial, additional protection required for the public health and safety. The Task Force recommendations presented in NUREG-0578 consisted of 23 specific requirements in 12 broad areas (nine in the area of design/analysis and three in the area of operations). They are all to be implemented in two stages by January 1981 in operating plants, plants under construction, and pending construction permit matters except for three items which involve rulemaking action. Two of these dealing with hydrogen were deferred to the long-term program. The other dealing with operation is being processed by the Office of Standards Development in rulemaking proceedings.

The ACRS considered the short-term recommendations on several occasions and issued a letter to the Chairman on August 13, 1979, indicating that the Committee agrees with the intent and substance of the Task Force recommendations. In addition the Committee indicated that a more flexible implementation schedule should be followed to more realistically give merit to certain operational situations such as timely refueling outages rather than some arbitrary date. The Task Force agreed to this recommendation. In addition the Committee recommended three additional instrumentation requirements for short-term action, i.e., containment pressure, containment water level, and containment hydrogen monitors. An additional requirement was added by NRC for remote capability for reactor coolant system venting of system high points.

The Office of Nuclear Reactor Regulation met with the Commission on September 6, 1979, to review the current licensing situation and outlined its proposed plan to proceed. Included in the plan were the overall short-term recommendations described above. Letters were sent on September 13, 1979, to the utilities discussing the short-term program as well as other required actions. These matters have been implemented on individual operating plants.

The short-term Task Force items are listed in the following table; however, there are other lessons learned that are being carried out by other Task Force Efforts. These include the Bulletin & Orders Task Force that deals mainly with the operating plants and the auxiliary feedwater system, the Emergency Preparedness Task Force dealing in the area of emergency planning,

SHORT-TERM TMI-2 ACTIONS
FOR ALL NUCLEAR POWER PLANTS
(NUREG-0578 et al)

Sect. No.	Action	Sect. No.	Action
2.1.1	Emergency Power Supply Requirement	2.1.8.c	Improved Iodine Instrumentation
2.1.2	Relief and Safety Valve Testing	2.1.9	Transient & Accident Analysis
2.1.3.a	Direct Indication of Valve Position	(ACRS)	Containment Pressure Monitor
2.1.3.b	Instrumentation for Inadequate Core Cooling	(ACRS)	Containment Water Level Monitor
2.1.4	Diverse Containment Isolation	(ACRS)	Containment Hydrogen Monitor
2.1.5.a	Dedicated H_2 Control Penetrations	(NRR)	RCS Venting
		2.2.1.a	Shift Supervisor Responsibilities
2.1.6.a	Systems Integrity for High Radioactivity		
		2.2.1.b	Shift Technical Advisor
2.1.6.b	Plant Shielding Review		
		2.2.1.c	Shift Turnover Procedures
2.1.7.a	Auto Initiation of Auxiliary Feed	2.2.2.a	Control Room Access Control
2.1.7.b	Auxiliary Feed Flow Indication	2.2.2.b	Onsite Technical Support Center
2.1.8.a	Post Accident Sampling		
		2.2.2.c	Onsite Operational Support Center
2.1.8.b	High Range Radiation Monitors		

particularly with respect to off-site preparations, and the Operating Training Task Force which is emphasizing better training in dealing with casualty-type situations by training with reactor simulators as well as improvements in the qualification program. In addition the industry is developing organizations to provide better training and evaluations capabilities for the operations groups; i.e., the Institute of Nuclear Power Operators and the Nuclear Safety Analysis Center.

In addition to the foregoing actions, a key lesson is that a better understanding and use of operating experience can be effective in improving the safety of nuclear plants. It is to be remembered that several precursor events took place on similar reactor plants prior to the TMI-2 accident. Although some preliminary studies of these events were performed, the full significance was not determined. A staff of experienced interdisciplinary people has been established whose sole job is to evaluate operating experiences and to ensure that the plant operators understand them and include such experiences into their training program and emergency procedures.

Other short term lessons learned actions include the development of an overall NRC Action Plan that covers those matters raised by the various review groups including the Presidential Commission and the NRC Special Inquiry Group. The plan will form the basis for establishing new additional licensing requirements for both the operating plants and near-term OL requirements. The new requirements for the operating plants deal with shift manning, licensing examinations, operating experience, B&O task force generic review items and control room habitability. New requirements for the near-term OL licenses include greater emphasis on the operating organization and management, an onsite safety engineering group, a review of control room designs, training for degraded core training, a review by the NSS vendor of emergency procedures and an NRR review of selected emergency operating procedures. In addition new requirements have been established for the preoperational start-up stage, i.e., training during low power testing and monitoring of power ascension testing.

The staff is currently engaged in improving the capabilities of its NRC operations Center at Bethesda, Maryland, in order to provide the Commission and senior staff members with vital plant parameters and information from licensed nuclear plants in the event of incidents or accidents. Improved capabilities which are under consideration for the center will include automatic data processing, data storage, data display and data recall capability to be achieved through the use of digital computers. This will enable the staff to monitor and evaluate the situation and potential hazard, advise licensees, and in an extreme case, to be able to issue orders governing such operations.

LONG-TERM PROGRAM

The requirements established for the short term are intended to address those matters where a short-term improvement in safety can be made. TMI-2 has raised a number of other significant questions and policy issues. These became the considerations for the long-term program.

The Task Force is completing its efforts for the long-term program that deal with the broader and more fundamental issues of reactor safety that emerged from the TMI-2 accident. The report of the Task Force dealing with the long-term aspects was published in October 1979. The long-term efforts are discussed in four areas: (1) Design Basis Accidents; (2) System Design Requirements; (3) Nuclear Power Plant Licensing; and (4) Nuclear Power Plant Operations.

DESIGN BASIS ACCIDENTS

The underlying philosophy of nuclear reactor safety is that protection against the release of radioactivity should not rely solely on one means of protection but requires multiple levels of protection, i.e., the concept of defense-in-depth. This concept has been implemented through the technique of specifying design basis events and associated acceptance criteria which conservatively assure that the desired levels of protection are attained. At Three Mile Island, the multiple levels of protection prevented the release of all but a small amount of radioactivity despite a number of equipment and human failures. However, the sequence of events at TMI included events such as operator error, unexpected system response, and extensive core damage, were beyond previously specified design basis events and violated current acceptance criteria. This does not necessarily indicate that the defense-in-depth concept is unsound. But the experience indicates to some the need to more seriously consider modifications of our criteria so as to extend the current design basis events to explicitly include significant degradation of core cooling, such as occurred at TMI, or perhaps even core meltdown, for some aspects of the design of nuclear power plants. In this regard, two specific changes to nuclear power plant design should be promptly considered and openly, perhaps, debated in a rulemaking framework. The first is the capability for containments to cope with the hydrogen gas generated by the metal-water reaction of a significant fraction (if not all) of the fuel cladding in a loss-of-coolant accident. The second is the capability for filtered venting of containments to ameliorate and delay the offsite consequences of a core meltdown by reducing the containment pressure peak for such an event. Such considerations are now being made for the Indian Point Units 2 and 3 and Zion Units 1 and 2 plants mainly because of their locations near highly populated regions, i.e., New York City and Chicago.

SYSTEM DESIGN REQUIREMENTS

The system design subgroup is reexamining the adequacy of current system design requirements. In examination of these requirements, the subgroup is considering modification of current requirements to include use of event tree, fault tree, and/or relative reliability methods to supplement the current deterministic licensing criteria. In addition, consideration is being given to methods to incorporate in the safety analysis operator action [inactive or error] and the role of operating procedures with relation to the system design requirements.

The subgroup is also evaluating the current system safety classification methods and is considering modifying these requirements to include additional systems in the safety grade classification as well as developing other system safety classifications. One classification system being considered is based on identifying systems important to safety, establish a rank of their order of importance and developing design requirements and criteria for various classifications. Recent operating experiences are showing the effects of failure of nonsafety grade types of equipment and the resulting challenges to plant safety features.

NUCLEAR POWER PLANT LICENSING

The Lessons Learned Task Force considered several specific topics within the general framework of how the NRC carries out its licensing activities. The areas in which recommendations include backfitting criteria, NRR organizational concepts and objectives, NRR emergency preparedness, and NRR evaluation and application of operating experience. With respect to backfitting a proposal was made for definitive criteria based on a required level of safety be articulated in the regulations and that the NRC finally put into its regulations that we require more from plants than the minimum requirements to meet the regulations. Organizationally the desirability of an integrated, interdisciplinary review team approach and added emphasis on operational safety aspects were emphasized. Our recommendations on emergency response addressed both the informational needs required as input and provision for a rapid NRR response and evaluation capability.

NUCLEAR POWER PLANT OPERATIONS

The Lessons Learned Task Force provided recommendations in a number of areas. A review of human factors in all operating control rooms has been recommended that would identify needed improvements in plant status assessment, improvements in safety system status monitoring, improvements in control and instrumentation hardware and reassessment of the number of required operator actions. In addition, it was recommended that the reactor operating experience evaluation programs that was recently required of all utilities be tied into a nationwide network for evaluation of reactor operating experience.

Recommendations for personnel included the recognition of present efforts underway by the industry's recently announced Institute for Nuclear Power Operations. We will emphasize improvements in training of nonlicensed operating plant personnel and independent verification of qualifications of nonlicensed operating plant personnel. The need for position task analysis and clearer definition of acceptable training programs for operating plant personnel will also be discussed.

SUMMARY AND STATUS

The staff is presently developing an overall NRC Action Plan that will incorporate all significant recommendations regarding the lessons learned from the TMI-2 accidents. Varous inputs including the Kemeny [5] and Rogovin [6] reports will be given proper attention. The overall Plan will address four major areas: (1) Operational Safety; (2) Siting and Design; (3) Emergency Preparedness and Radiation Effects; and (4) NRC Organization, Management, Practices and Procedures. In conjunction with the development of the overall Action Plan, additional requirements for near-term operating licenses will also be specified. The staff will discuss the plan with ACRS and Commission to ensure proper promulgation and allocation of resources on a systematic basis. A resumption of licensing activities following the recent pause imposed by the NRC has taken place with the recent issuance of the Sequoyah low power operating license.

It is clear that nothing in the world of nuclear power generation will be the same as it was before March 28, 1979. The accident at Three Mile Island becomes a historical landmark, a watershed event whose worldwide technical, legal and societal implications are only now beginning to emerge. Safe and reliable operation of nuclear power plants goes beyond the acceptance of whatever the NRC requires. Clearly the responsibility rests with the industry and the utility to accommodate and respond to the lessons learned from the TMI-2 accident. It is also important that we follow-up on any significant experiences that bear on assessing the effectiveness of the ensuing lessons learned actions, the February 26, 1980 event at the Crystal River 3 facility.

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TMI 2 IMPACT ON THE FRENCH NUCLEAR PROGRAM AND SAFETY ANALYSIS

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ABSTRACT

Almost immediately after the TMI accident, Electricité de France (EdF), Framatome and the French safety authorities started a large-scale program of actions designed to analyse and understand the causes of the accident, and draw lessons applicable in France.

This paper discusses these actions and the main conclusions of TMI accident analysis in France, notably :
- the fundemental role of plant operators, and the importance of operator training, written instructions and procedures, and diagnostic aids,
- the importance of feeding back operating experience to design teams, and incorporating the results of accident and post-accident studies in operating procedures,
- the necessity to improve knowledge of core cooling modes, including during two-phase flow and natural circulation,
- measures to improve particular systems and components.

1 - INTRODUCTION

French analysis of the Three Mile Island accident and its application to the French nuclear program have been considerably facilitated by :
- the attitude of the US organizations concerned, notably the NRC and industry (particularly the EPRI), which have widely distributed their accident analyses and findings, both preliminary and final,
- the fact that in France there is only one plant owner and operator (EdF) and only one NSSS supplier for PWRs (Framatome), and the high degree of standardization of French plants (2 standard plant designs - 900 and 1300 MWe - each represented by a large number of identical plants), have limited the number and extent of TMI-related studies, and allowed rapid implementation of improvements inspired by the accident.

From a technical point of view, the impact of TMI on French PWR plants differ little from those applied in the US. This paper therfore discusses how EdF and the French safety authorities have handled the problems raised by the accident, and the most significant actions undertaken.

2 - INITIAL ACTIONS

On April 18th, 1979, the Ministry of Industry decided to reinforce surveillance by nuclear installation inspectors at operating power plants. The Ministry also addressed a letter to EdF requesting the latter to communicate within one month the initial lessons for French PWR plants, based on the NRC's preliminary accident analysis. This letter also contained a list of questions on various technical aspects directly related to knowledge available at that date on the TMI sequence of events.

In reply, EdF submitted on the 18th of May, 1979, its first general conclusions on the TMI accident and a list of suggested objectives.

These conclusions are that existing PWR safety analysis and design methods based on the concept of defense in depth and use of Design Basis Accidents have not been invalidated by the TMI accident, but that greater attention must be paid to incidents of moderate frequency which can initiate accidents. This involves enhancing the reliability of equipment capable of causing such incidents, improving incident diagnosis, and more clearly defining necessary operator actions.

Based on these conclusions, the initial objectives proposed by EdF included :

- improve some calculational models, and continue post-accident studies for design basis accidents,
- determine any deficiencies in system design, and take appropriate actions,
- determine appropriate technological improvements for equipment capable of initiating incidents,
- facilitate operator action by providing better diagnostic aids and clearly-written instructions for each type of incident.

3 - FINDINGS OF THE "GROUPE PERMANENT"

The French "groupe permanent" met during four days in June and July 1979, to study the TMI accident, based on EdF's above-mentioned document, and a report prepared by the Institut de Protection et de Sûreté Nucléaire (IPSN), which made use of information from French experts sent to the US to study the accident. Following these meetings, the "groupe permanent" communicated its findings to the Service Central de Sûreté des Installations Nucléaires (SCSIN). These findings included comments on the accident, with particular emphasis on factors which appeared to have a determining influence, and two series of recommendations concerning respectively safety analysis and measures to be taken or studied for French PWR plants.

For the "groupe permanent", the fundemental cause of the seriousness of the TMI accident is that during the first hours following accident initiation, operators did not, correctly diagnose the nature of the accident, nor foresee the situation that later arose, and thus took inappropriate action. The "groupe permanent" concluded that safety analysis should in the future pay more attention to difficulties encountered during operation of reactors, particularly the necessary consistency between Engineered Safety Features designed to handle accidents and the rules and procedures applied by plant operators when an accident occurs. For this reason, the "groupe permanent" first series of recommendations concerned :

- verification of the consistency of operating instructions with the accident studies performed during plant design and with post-accident studies. Within this context, it is necessary to ensure that operators are provided with unambiguous indications permitting diagnosis of the actual condition of the plant, particularly core cooling,
- improvement of knowledge on certain modes of natural and forced circulation core cooling for the types of break that can affect the primary cooling system, including during periods when cooling is by the residual heat removal system,
- the necessity of in-depth analysis of the main incidents at operating plants, as this can provide early indication of more serious potential accidents.

The "groupe permanent" also confirmed its interest in EdF's ongoing studies on failure of redundant systems (ATWS, total loss of power supply, loss of ultimate heat sink, loss of feedwater supply to steam generators).

The second series of "groupe permanent" recommendations, which cannot be detailed here, concerned technical studies and suggested modifications relative to particular aspects of reactor design, instrumentation and operation.

On the 3rd of August, 1979, the SCSIN addressed a letter to EdF reiterating most of the "groupe permanent"'s recommendations, and announcing the creation under its responsibility of a working group comprising representatives from EdF, Framatome and the safety authorities, to study in detail problems related to the interface between reactor design and operation (cf. § 6).

4 - ACTIONS UNDERTAKEN BY EdF

In August 1979, EdF established a detailed program of actions to be continued or initiated. This program was based on EdF's own analysis of the situation, which had been developed between May and July 1979, and on the SCSIN's requests in its letter dated August 3rd, 1979.

The program comprises 46 actions, divided into 185 sub-actions each pertaining to a study or design change. The 46 actions may be grouped into three categories : system studies, equipment improvement, and increasing the reliability of operator actions.

4-1. System studies

These have 4 objectives :

a) Modification of system operating parameters, so that less use is made during normal transients and operating incidents of components whose failure could result in an accident.

Four actions (n° 9, 13, 38 and 42) pertain to this objective. They involve studies on maintaining normal feedwater supply to steam generators after reactor scram ; improvement of solid RCS pressure control and instrumentation ; narrowing the range of temperatures over which the RCS is solid ; adapted control of pressurizer relief valves and condenser by-pass valves.

b) Improved knowledge of accident situations : This involves two actions (n° 10 and 32). They mainly concern evaluation of incondensable gas volumes ; studies of small and medium size RCS breaks ; study of residual heat removal system breaks.

Some of these actions are covered by ongoing R&D studies, certain of which were starded 2 years ago and will end in 1 or 2 years time.

c) Study of incidents of moderate frequency, which can be the origin of more serious accidents. The incidents in question are those postulated during design, plus additional incidents encountered during operation. Plant testing and operation can reveal deficiencies in design, and it is most useful to study situations resulting from failures at these points, with a feedback system to fully inform design teams of relevant experience acquired during testing and operation. In this respect, EdF has complete its internal organization to take better benefit from experience of operating reactors. An evaluation of main incidents will be submitted half yearly by the licencee to Safety Authorities. This evaluation will include : analysis of the incident, lessons to be learned and implementation of improvements on other plants. This periodic evaluation should make easier the determination of forerunner events.

This objective involves two major actions (n° 33 and 40).

d) Assessment of post-accident radioactive product confinement, and determination of appropriate modifications, including any necessary improvements to systems operative during post-accident situations.

Sixteen actions (n° 1 to 8, 10 to 12, 18, 22 to 24, and 41) are related to this objective. They mainly concern assessment of rooms and systems involved in waste handling and ventilation in the Nuclear Auxiliary, fuel, connecting and ESF buildings ; analysis of the feasibility of storing liquid wastes in the reactor building ; use of hydrogen recombiners ; evaluation of Engineered Safety Systems such as the auxiliary feedwater system, containment isolation system, and safety injection system.

4.2. - Equipment improvement

The matter is to verify that safety-related equipment are able to perform correctly in actual operating conditions, and then to make any necessary modifications.

This includes qualification testing, and monitoring the in-service behaviour of equipment, to detect any deficiencies and permit appropriate remedial measures.

Five actions (n° 14, 16, 17, 38 and 42) are being continued or have been engaged in this respect. They mainly concern equipment installed inside the containment, and which must operate under post-accident conditions, particularly pressurizer relief and safety valves, residual heat removal system valves, and leaktightness under accident conditions of safety injection and containment spray circuits inside and outside the containment.

4-3. Increasing the reliability of operator actions

This objective involves verifying that :

a) Operators are adequately trained. Training includes work on simulators, which provides good knowledge of plant behaviour during incidents. It is envisaged to also use simulators to provide hands-on experience with typical accidents (action 36).

b) Information made available to operators is reliable, and appropriately displayed, with an adequate priority-indicating method. It has been decided to study improvements to display of alarm conditions, core temperatures, and status of Engineered Safety equipment and safety-related valves (actions 19, 20 to 22, 37, 39 and 46).

c) Operating instructions are clear and easy to apply, correctly related to control room information displays, and consistent with the results of accident and post-accident studies. A major effort is being undertaken in association with the NSSS supplier to simplify instructions and make them easier to use.

d) In-service O&M procedures are correctly applied and limit the risks of human error. Practical application of these procedures will be monitored in the field, with initial findings issued at first refueling of the Tricastin plant (action 44).

e) Emergency plans are adequate, with examination of the usefulness of constituting a team of experts to assist the plant manager in the event of an accident (actions 34, 35 and 45).

5 - PRESENT PROGRESS ON ACTIONS

As of March 15th, 1980, about half the actions involving studies had been the subject of reports submitted by EdF to the safety authorities.

Also, a first series of design changes had been completed or were underway at this date. The most important are :

- for the Fessenheim and Bugey plants, cancellation of safety injection startup upon simultaneous low pressurizer pressure and low pressurizer level,
- provision of remote control and actuation of iodine filters in Nuclear Auxiliary Building ventilation circuits,
- increased indication range for thermocouples measuring the coolant temperature at core outlet,
- installation of saturation pressure margin indicators,
- installation of automatic circuits for closure of pressurizer block valves when low RCS pressure is indicated,
- implementation of improvements on safety valves of the residual heat removal system.

Moreover, actions previously undertaken to qualify safety-related active components (especially the valves of RCS) were greatly enhanced after the TMI accident.

6 - WORKING GROUP ON PLANT DESIGN-OPERATION INTERFACING

The role of this working group is very important, as it concerns a key problem highlighted by the TMI accident, namely the interface between plant design and plant operation.

The group started work in October 1979, using documents stemming mainly from the EdF actions mentioned in § 4.3.b and c above. At the time of writing, the working group has set itself 8 tasks :

1) Review consistency of automatic actions and abnormal and emergency procedures with the results of accident studies ; review internal consistency and validity of these procedures.

2) Review consistency of automatic actions and abnormal and emergency procedures with the results of post-accident studies.

3) Review consistency of automatic actions and abnormal and emergency procedures with information made available to operators ; examine the real significance of measured parameters and the reliability of information displayed in the control room.

4) Verify that information made available to operators enables unambiguous diagnosis before following instructions involving inhibition of an Engineered Safety Feature.

5) Analyse and substantiate differences between the abnormal and emergency procedures at different plant units.

6) Study problems related to unavailability of safety-related equipment ; revise technical operating specifications influencing availability.

7) Examine all credible core cooling conditions, analyse their stability and changes from one condition to another, and examine representative parameters. The purpose here is to confirm that all possible core cooling conditions can be characterized by a limited number of physical parameters, and that it is not necessary to perform studies on a great many plausible accident sequences.

8) Study the type of information and diagnostic aids made available to operators. This long-term task involves examining the design of control rooms, and considering more extensive use of automatic circuits and computers in plant operation and for diagnosis.

As of mid-March 1980, the 10 sets of emergency procedures had been reviewed in detail. No major inconsistencies had been found with the results of accident and post-accident studies. However, this review demonstrated the necessity of completely rewriting these procedures, so as to improve accident diagnosis based on control room indications, improve criteria authorizing operators to implement safety actions during an accident, and make instructions easier to use. This work has been started with the assistance of specialists on human behaviour, and should be completed by the end of 1980.

7 - CONCLUSION

In view of the size of the French nuclear program, a very considerable amount of work was necessary by both EdF and the safety authorities to rapidly analyse the TMI accident and draw conclusions applicable in France. With a few minor exceptions, for which remedial action was rapidly taken, this revealed no design deficiencies justifying interruption of the French program or delays in commissioning of plants. However, analysis did reveal the absolute necessity of giving increased attention to certain aspects of plant operation, including improvement of all aspects designed to permit operators to respond correctly in the event of an accident.

This improvement work is well underway, and should be rapidly completed, thanks in particular to operating experience with plants already in service.

ALTERNATIVE EVENT SEQUENCES
OR
WHAT MORE COULD HAVE GONE WRONG?

by

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ABSTRACT

The sequence of events in the TMI-2 accident was examined to determine if one or more equipment malfunction or operator action could have changed the consequences significantly. Most such scenarios were benign, but a couple were explored whose consequences were more difficult to calculate. To assume a conservative evaluation, conditions leading unequivocally to fuel melting were postulated and the subsequent events examined. It was found that for this accident, the integrity of containment would not have been successfully challenged by a steam explosion, over-pressure, or by penetration of the concrete by molten fuel. An associated conclusion was that off-site releases of iodine would not have been changed significantly.

Very nearly all discussions of the TMI-2 accident touch upon the subject of various possible sequences of events or scenarios that might have developed, starting with the actual situation and leading one way or another, from the actual situation to a variety of results--some more, some less severe than the actual accident. Most of these alternative paths[1] are either benign (i.e., closing the PORV) or worsen the accident to a minor degree (i.e., filters in worse condition). However, at a time of about 3 hours into the accident, the reactor vessel was becoming very short of water; and if the high pressure injection (HPI) system had not been restored to full flow, fuel certainly would have reached the fusion temperature and some would have melted before the HPI came on automatically at 3 hours and 55 minutes. Because of the uncertainty associated with the consequences of not turning on the HPI at 200 minutes, the Staff of the President's Commission chose to postulate* a fuel melting condition at a time into the accident of about 3 to 4 hours.

The objective of the exercise was to investigate the consequences of such an event for the TMI-2 accident, and, in particular, to determine if the containment would fail and if the release of fission products to the environment would have been changed significantly.

The postulate to initiate this arbitrary sequence of events was that the operators did not restart the HPI at 3 hours and 20 minutes and did not allow any cooling of the core by water after this time except for the core flood tanks. The postulated sequence of events and consequences to the containment and environment will be discussed briefly.

1. Time to reach melting temperatures: At 200 minutes, the HPI was not restarted; fuel temperatures were rising and would reach melting temperatures at some point in the core in less than an hour, possibly only minutes, depending on the detailed assumptions.

2. Fuel melting: Some fraction of the core was calculated to melt and reach the lower plenum in the reactor vessel where it would release heat to the water. The time for melting could take as little as an hour but could be much longer, depending on steam flow and efficacy of thermal radiative cooling. No proof was offered that a large fraction of the core melts; the fraction might be small.

3. Steam explosions in the reactor vessel: There would be enough energy stored in molten fuel such that if highly efficient transfer of this energy to water were to occur upon contact, it is conceivable that the explosive force of rapidly generated steam could cause rupture to the pressure vessel and threaten the containment. As a practical matter, however, it is difficult to postulate physical mechanisms which could permit highly effective energy transfer from large quantities of fuel to water. Further, it is difficult to imagine how large quantities of fuel and water could be caused to interact simultaneously, since a sufficient quantity of fuel would very likely not melt at the same instant, nor would a sufficient quantity of molten fuel in small particles all contact the water at the same instant. In addition, recent reactor safety

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*The TMI-2 core was flooded with water by operator initiation of the HPI system at 200 minutes. To continue to deny an remedial measures by the operators is very conservative and unrealistic. Such measures would be possible well into thus hypothetical fuel melting accident, but these were denied in order to investigate consequences to the extent possible in the time available.

experiments directed at resolution of the steam explosion potential indicate that mechanisms for efficient interactions are not found. In effect, viewed as a steam engine to do work, the mechanism of dropping hot metal into water is inefficient and undependable. The conclusion was reached that a steam explosion in the pressure vessel would not rupture the vessel or the containment building. More recent investigations (within the past few months) support the conclusions reached in the autumn of 1979. Given the continuous delivery of molten fuel to the water in the lower plenum, an additional quantity of steam would be formed. Under some conditions, this steam can provide a heat removal capability equal to or greater than the decay heat rate.

4. Debris bed cooling: When the molten fuel falls into the water in the lower plenum, cooling and fragmentation is expected even if no steam explosion is caused. Debris beds of certain particle sizes, not too large or small, can be cooled by water under high pressure conditions. However, because no believable predictive model is known and to continue the conservative nature of the study, the assumption was taken that the fuel would form a molten pool sooner or later.

5. Penetration of reactor vessel: Given the existence of a large amount of molten fuel in the vessel, penetration was predicted to occur in a relatively short time, some tens of minutes. The amount of molten fuel that could accumulate in the lower plenum is nearly unknown because of uncertainties mentioned earlier. Proof of a large amount was not offered.

6. Containment pressure: Given failure of the reactor vessel and escape of steam and hot fuel, the pressure in the containment was evaluated. All of the latent and sensible heat in a mass of fuel equal to that of the whole core was placed into the vaporization of water and added to the pressure already present. The total pressure evaluated by this conservative method was less than that in the design basis accident. At this point, it should be clear that the postulated, extended accident is specialized to the TMI-2 conditions; the extended accident was begun at a time when the containment pressure was low and much of the primary system heat had been quenched. Other reactors and different accidents might also survive this step, but a different analysis would be necessary.

7. Fuel reaching the cavity below the reactor vessel: The matter of steam explosions was considered a second time. Because cavity leakage paths exist, consequences of the interactions between fuel and water in this area were less serious than in the vessel. In addition, the amount of water in the cavity at this time was believed to be insufficient for the purpose of producing significant steam explosions.

8. Penetration of the containment basemat: The action of hot fuel on a containment floor is to melt, erode, and disintegrate the reinforced concrete. Steel reinforcing and metallic oxides in the concrete would dissolve in the molten uranium dioxide. Water vapor and carbon dioxide would be liberated during the interactions and reach the containment air space either through or around the molten material. Penetration of a basemat was predicted in WASH-1400 to require 18(+10,-5) hours but research, both experimental and theoretical since 1975, has extended the extimate of time considerably. Estimates in Germany have ranged up to 13.5 days for 20 feet of concrete while estimates in the United States range between 3 days and never for a 13-ft thick basemat. It has become clear that solidification would occur in about 2 days, well before the minimum time predicted for penetration and thus will change the physical processes involved. Should the basemat be penetrated, the material at Three Mile Island would have encountered solid siltstone, which is essentially impervious to water and gases. The containment air pressure would be above atmospheric because of the addition of water vapor and gases from the decomposition of concrete, but the design pressure would not be exceeded.

9. Containment failure: The three mechanisms that might cause failure of containment--projectiles from a steam explosion, overpressure, and penetration of the basemat--have been examined. The conclusion was reached that containment would not fail and result in an uncontrolled escape of fission products to the atmosphere; the amount escaping would be less than that in the design basis accident.

10. Fission product behavior: The possible fission product inventories in the primary system, the containment building, the auxiliary building, and the environment were estimated by analogy to the conditions that actually existed subsequent to the accident at Three Mile Island. During the accident between 2.5 and 15 million curies of krypton and xenon, primarily Xe-133, were released to the environment, but only 15 curies of iodine-131 and, to our knowledge, no cesium, strontium, or other non-volatile species. The remarkable success in regard to iodine is believed to have been achieved because of the chemical reducing conditions existing at the point of release from the fuel and because of the water or steam environment. The iodine went into solution as an iodide ion and remained so, becoming even more firmly fixed when the containment sprays increased the pH by injecting sodium hydroxide into the containment.

The leakage path from the containment is believed to have been by way of the letdown line, a water pathway to tanks in the Auxiliary building. Any non-volatile fission products, e.g., cesium-137, carried in this piping system would be at such a low temperature that evaporation would not occur. However, the noble gases, krypton and xenon, would readily leave the water solution and leaks in the header system connecting the tanks allowed xenon to escape to the Auxiliary building and later to the environment. Creation of methyl iodine is usually postulated to be a significant fraction of the inventory of iodine, but none is expected to have been formed in the primary system and, once in solution, creation of this molecule is unlikely.

The fuel melting accident was postulated as an extension of the actual accident that occurred. Thus, the chemical conditions existing in the core would be expected to be chemically reducing, as actually existed. The iodine would go into solution and remain so. More extended operation of the containment sprays could be expected, thus fixing the iodine in solution even more strongly. More iodine would have been released to the containment, but the amount escaping to the environment should not be increased by more than a relatively small factor.

The escape of non-volatile fission products (cesium, rubidium) to the Auxiliary building would be increased in proportion to the escape from the fuel, but no reason could be found to postulate release to the environment.

If most of the fuel should melt, as was required by the postulates, most of the noble gases would escape from the fuel. If the escape fraction from fuel in the TMI-2 accident was 50%, the escape of noble gases to the environment could be increased by a factor of about two. This enhanced release to the environment would then have caused about 4000-5000 person-rem instead of about the 2000 observed. A second source of leakage to the environment can be identified with high pressures in the containment building. Analyses in, for example, NRC Safety Evaluation reports postulate a fixed percentage of the containment volume for a period of about a day after an accident. This second, non-mechanistic source of leakage to the environment cannot be evaluated quantitatively. However, no reason could be found to require this leakage to be anything but much less than that postulated in standard design basis accident analyses because containment pressure was relatively low.

SUMMARY

An analyses of alternate Event Sequences in the TMI-2 accident led to an uncertain situation relative to whether or not some fuel would melt. In order to bound the situation, a scenario was postulated in the context of the TMI-2 accident that would unquestionably lead to fuel melting. Mitigating conditions in the core were denied except for possible action of the core flood tanks. It was found that the TMI-2 containment would not have breached to the environment by excessive pressure, by a steam explosion, or by penetration of the base of the building by the action of molten fuel. Because the accident was an extension of the TMI-2 accident, the fission products (especially iodine and cesium) escaping from the fuel would encounter essentially the same conditions that existed in the actual accident. Thus, there is no reason to assume that the amount of iodine or cesium escaping to the environment would have been changed by a large factor even though more of each would have been released to the primary system. The noble gases, xenon and krypton, would have escaped very nearly completely from the fuel, and the off-site escape could have been increased by about a factor of two.

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CALCULATIONS OF HEAT REMOVAL DUE TO NATURAL CONVECTION IN THE TMI CORE WITH POSTULATED DAMAGE

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ABSTRACT

Calculations are presented of the temperature distribution inside the damaged reactor core at Metropolitan Edison's Three Mile Island nuclear power plant. These results were obtained after the breakdown occurred at a time during reactor shut-down when one of the reactor core coolant pumps was still operating. Their purpose was to determine whether the temperature of the liquid within the core would reach or exceed the saturation temperature when the final core coolant pump was turned off. Our conclusion, arrived at and reported before the pump was actually stopped and based upon assumptions about the extent and type of damage to the fuel rod assembly within the core, was that the fluid temperature would rise significantly, but that boiling would not occur. Events at Three Mile Island confirmed these predictions.

INTRODUCTION

This work describes a quick response, cooperative effort between the Electric Power Research Institute and JAYCOR to estimate the fluid flow and heat transfer in the damaged TMI core for the purposes of helping the Nuclear Regulatory Commission (NRC) decide on a course of action. At the time only one recirculation pump was operating to keep coolant moving through the core and there was concern that the pump might eventually fail due to radiation exposure, as had much of the instrumentation. Furthermore, the outflow temperature was considerably higher over the central part of the core suggesting internal blockage. It was critical to know whether the reactor would cool itself through natural convection between the hot core and the steam generator or whether other emergency equipment, outside the containment, might have to be activated. If it was reasonably certain that natural circulation would be sufficient, the NRC preferred to shut down the pump while the instrumentation could still monitor the process. A two-week deadline had been set for collecting input on the possible consequences.

PROBLEM DEFINITION

A possible scenario for the first few hours of the TMI event was that the top one-third of the core had been uncovered, damaging the rods and leaving the fuel pellets in rubble piles on the upper three positioning screens. An idealized configuration of the core, therefore, was assumed to be three semi-permiable rubble disks, one above the other, occupying the center half of the flow area and separated by unobstructed gaps. The remainder of the rods were assumed to be intact. Based on decay heat calculations and confirmed by the measured heat balance, the core was producing 3.5 MW thermal. These calculations also provided an estimate for the spatial distribution of power density. Loop calculations were used to estimate the overall circulation through the system due to cooling in the steam generator. Those calculations showed that without blockage the flow would be sufficient for cooling. The problem posed of this investigation was: Given the assumed blockage geometry, heat generation, and flow rate would natural convection between the disks be sufficient to remove the heat and prevent local boiling?

TECHNICAL APPROACH

The problem required obtaining detailed heat transfer due to flow driven by forced convection and buoyancy through a complex and possibly porous geometry with irregular heat addition. It was decided to modify JAYCOR's Equation Independent Time-Average Conformal Coordinate (EITACC) computer code. EITACC has been developed for making accurate, transient calculations of fluid flow in a variable geometry with emphasis on a structured, easily modified architecture. It was modified to handle a fluid temperature field with heat addition, semi-permiable blockages, and the required inlet and outlet boundary conditions. The code was then used to solve the time dependent mass, momentum, and energy conservation equations in a two-dimensional cylindrical geometry with blockage.

Physical Model

A side view schematic of the reactor core is shown in Figure 1. The vertical lines represent those regions where the fuel rods are assumed to have remained intact. We make the following assumptions on the geometry as it affects the nature of the flow.

(a) The reactor core and power distribution is axisymmetric. This allows us model the flow as two-dimensional.

(b) The screens are porous. The velocity of the fluid through the screen is proportional to the pressure drop,

$$v = \gamma \frac{\partial p}{\partial z} \; . \tag{1}$$

Setting γ to zero makes the screen impenetrable.

(c) The power produced by the debris is totally transmitted to a thin layer (one finite difference cell width) of water just above the screen. In those sections of the core which are assumed undamaged, the distribution

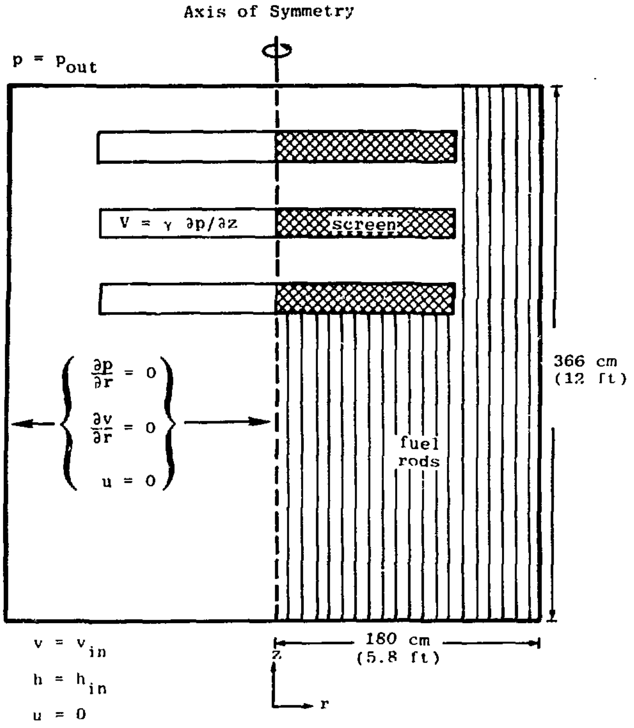

Figure 1. Reactor Core Schematic Showing Boundary Conditions

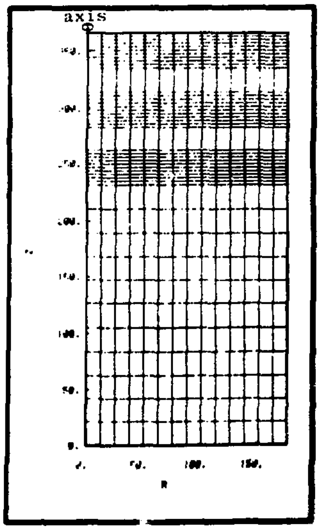

Figure 2. Finite Difference Mesh Used in Three Mile Island Calculations (14 x 43)

of power production (Applied Power Density) at a point within the core is given by:

$$\text{APD}\ (r,z) = \frac{P_T}{\pi R^2 z}\left[2.458 - 1.365\left(\frac{r}{R}\right)^2\right] \left[\frac{z}{Z}\left(1 - \frac{z}{Z}\right)\right]^{0.3}\ \text{cal/sec/cm}^3 \tag{2}$$

$R = 180$ cm
$Z = 366$ cm

where P_T is the total power produced by the reactor. In the present case the reactor is essentially shut down, and is producing only about 3.5 megawatts of power. The power produced by the debris is assumed to be the power produced by the rods (before crumbling) between the screen just below and the one just above the debris.

(d) The flow resistance due to the intact fuel rods is modeled by friction forces in the equations for conservation of momentum. These forces are proportional to the square of the velocity and have been calibrated by the manufacturer in other analyses.

(e) The inlet velocity and inlet heat are assumed to be uniform.

Fluid Flow Model

The dominant phenomena we wish to model is the natural heat convection through the reactor core. Since the flow is slow, buoyancy effects due to density variations in the fluid, which are in turn dependent upon temperature variations, is one of the driving forces. However, because the change in density is small compared to the density itself, we can use the Boussinesq approximation and neglect the density variation in the momentum transport (advection) terms and the continuity equation. This leads to the following system of equations for the fluid flow within the reactor core:

$$\frac{1}{r}\frac{\partial ru}{\partial r} + \frac{\partial v}{\partial z} = 0 \tag{3}$$

$$\frac{\partial u}{\partial t} + \frac{1}{r}\frac{\partial ru^2}{\partial r} + \frac{\partial uv}{\partial z} + \frac{1}{\rho_\ell}\frac{\partial p}{\partial r} = \nu\left(\nabla^2 u - \frac{u}{r^2}\right) - \alpha|u|u \tag{4a}$$

$$\frac{\partial v}{\partial t} + \frac{1}{r}\frac{\partial ruv}{\partial r} + \frac{\partial v^2}{\partial z} + \frac{1}{\rho_\ell}\frac{\partial p}{\partial z} = \nu\ \nabla^2 v + \frac{\rho}{\rho_\ell}g - \beta\ v|v| \tag{4b}$$

$$\frac{\partial h}{\partial t} + \frac{1}{r}\frac{\partial ruh}{\partial r} + \frac{\partial vh}{\partial z} = \mu\nabla^2 h + s \tag{5}$$

$$T = h/(\rho c_p) \tag{6}$$

$$\rho = \rho_\ell[1 + \varepsilon(T_s - T)] \tag{7}$$

where for any scalar ϕ:

$$\nabla^2\phi = \frac{1}{r}\frac{\partial}{\partial r}\left(r\frac{\partial\phi}{\partial r}\right) + \frac{\partial^2\phi}{\partial z^2}$$

In the above equations $U = (u,v)$ is the velocity in the axisymmetric cylindrical coordinate system (r,z) with radial r and vertical z directions, t is time, p is pressure, h is heat, ρ is density, ρ_ℓ is the ambient constant density, T is temperatuare, c_p is the specific heat ratio, T_s is the saturation temperature, α is the radial fuel rod friction coefficient, β is the vertical friction coefficient, s is the heat source which is found from the Applied Power Density function (Equation 1), and μ and ν are the diffusion coefficients for the momentum and heat equations, respectively.

The boundary conditions on this set of equations are given in Figure 1.

Numerical Method

The scheme uses a transient relaxation to the steady state beginning at some initial (guessed) flow field. Figure 2 shows the finite difference mesh used in all the calculations. The resolution of this mesh is highest in the areas between the screens to provide for adequate resolution of the temperature and flow in these regions. The finite difference scheme is a composite of time averaging for transient terms, upwind differencing for advection terms and central differencing for diffusion terms. The details of the numerical method are for the most part given in Reference [1].

Difficulties Encountered

The time scale for natural convection to become established and for the system to approach a new steady state after the pump was turned off was much greater than the time step allowed by the numerical solution algorithm to resolve the detailed flow. A detailed calculation of the entire transient would have been too costly. Therefore, a direct solver was developed to obtain the steady state temperature field for any particular flow pattern and used to accelerate the approach to steady state. This is accomplished by solving

$$\frac{1}{r}\frac{\partial ruh'}{\partial r} + \frac{\partial vh'}{\partial z} - \mu\nabla^2 h' = s$$

for the "steady state" heat distribution using a direct matrix solution technique with u and v taken from the hydrodynamic calculation and treated as fixed, and then under-relaxing.

PREDICTED RESULTS

The primary contribution of this work is the demonstration of the importance of the internal flow within the reactor to the heat transfer from the core. When the final reactor core coolant pump was turned off the mass flow rate through the core was reduced to one per cent of its previous value. One might expect that the heat transferred to the fluid would then increase proportionally. However, due to the nature of the internal flow found by these calculations, we predicted a temperature increase over the inlet temperature of only a factor of two at the hottest point in the reactor core as is shown in Figure 3. The reason for this small increase is illustrated in Figure 4. At the high flow rate due to the pump, a recirculation zone is set up between the screens which hinders the heat transfer to the outlet. However at the low flow rate, the cooler inlet water passes over the screens and then into the outlet flow. Hence the slower flow is more efficient in removing heat from the rubble piles deposited on the screen.

The computer code EITACC was used to numerically solve the equation set (Equations 3 to 7) for the heat distribution and flow within the reactor core. A parameter study was conducted to learn the affect of inlet flow rate and residual reactor power on the maximum temperature occuring within the core. First, we established the temperature and flow within the core when one reactor coolant pump was operating. We then determined the temperature and flow when that pump was turned off. The accelerated steady-state results indicated a peak outlet temperature rise and a peak internal temperature value that was well below saturation. Because the thermal output of the core was slowly decreasing and because of the uncertainty of the estimated flow rate due to steam generator cooling, a set of runs varying these parameters was made. These results are summarized in Table I. All cases indicated that boiling would not occur. The maximum fluid temperature given in Table I always occurs at a position within the reactor core near the centerline (axis of symmetry) just above one of the three positioning screens. The inlet mass flow rate into the core with one pump running was taken to be 0.57 m/s (1970 lbm/sec). The flow rate with all pumps shut down, which takes into account the natural convection due to the steam generator, was assumed to be one per cent of that value or 0.57 cm/s. We considered two rates of power production, 2.5 and 3.5 megawatts.

Although constraints on computer time prevented us from following a full transient to steady state we did study a portion of such a transient. Figure 3 shows the change in maximum temperature when the one remaining reactor coolant pump is shut down in a manner such that the inlet flow rate decreases from 0.57 m/s to 0.57 cm/s uniformly over a period of 2.5 seconds. The reactor core in this case is producing 3.5 megawatts power. Time zero refers to steady state for the high flow rate. The fluid temperature in the core does not increase immediately when the pump is shut down but instead remains relatively constant for a few seconds and then decreases rapidly before leveling off and slowly begins ascending. The steady-state value is shown in the upper righthand corner of Figure 3. The drop is associated with a transient flow readjustment due to too abruptly stopping the pump in our calculations, not an overall cooling effect. In the actual event the pump coasted for two to four minutes after it was tripped. Figure 4 shows streamlines and contour plots of temperature at various times during the transient.

Table I. Maximum Fluid Temperature at Steady-State for Various Flow Rates and Powers

Case		Maximum Temperature	
Inlet Flow Rate	Power	Increase Over Inlet	Absolute
57.0 cm/sec (1970 lbm/sec)	3.5 MW	21°C (38°F)	420°K (297°F)
57.0 cm/sec (1970 lbm/sec)	2.5 MW	4°C (7°F)	403°K (266°F)
0.57 cm/sec (19.7 lbm/sec)	0.6 MW	14°C (25°F)	413°K (284°F)
0.57 cm/sec (19.7 lbm/sec)	1.25 MW	22°C (40°F)	422°K (300°F)
0.57 cm/sec (19.7 lbm/sec)	2.5 MW	35°C (63°F)	444°F (340°F)
0.57 cm/sec (19.7 lbm/sec)	3.5 MW	46°C (83°F)	453°F (356°F)
0.28 cm/sec (9.8 lbm/sec)	2.5 MW	53°C (95°F)	460°F (369°F)
0.57 cm/sec (19.7 lbm/sec)	2.5 MW	35°C (63°F)	444°K (340°F)
1.14 cm/sec (39.4 lbm/sec)	2.5 MW	29°C (52°F)	430°K (315°F)
57.0 cm/sec (1970 lbm/sec)	2.5 MW	4°C (70°F)	403°K (266°F)
Saturation temperature = 557°K (900 psi,			

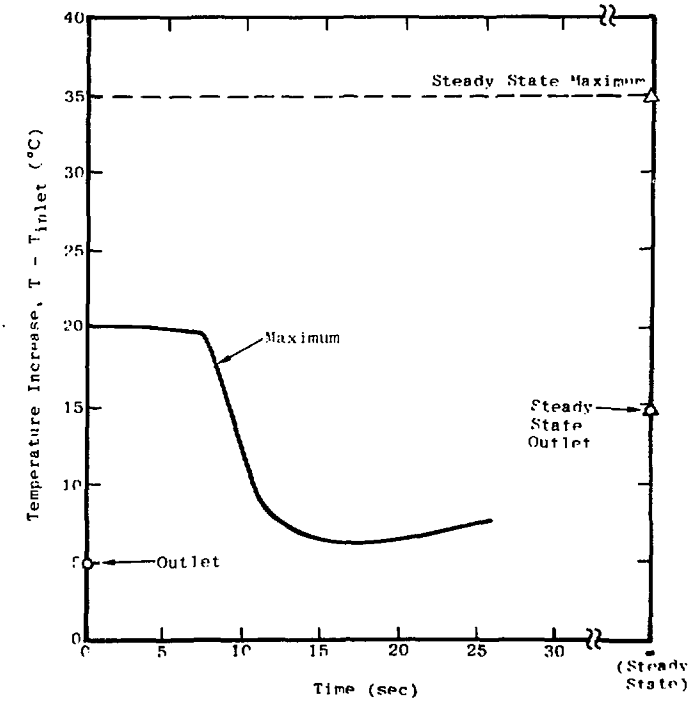

Figure 3. Over-Temperature Transient for Coolant Pump Shutdown Over 2.5 Seconds

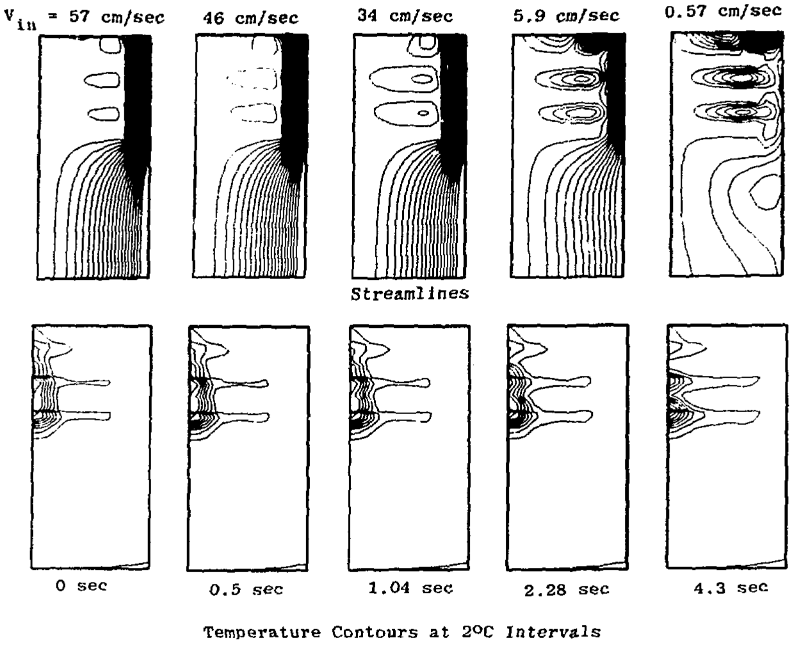

Figure 4a. Flow Transient During Pump Shut Down Over 2.5 Seconds at 3.5 MW Power

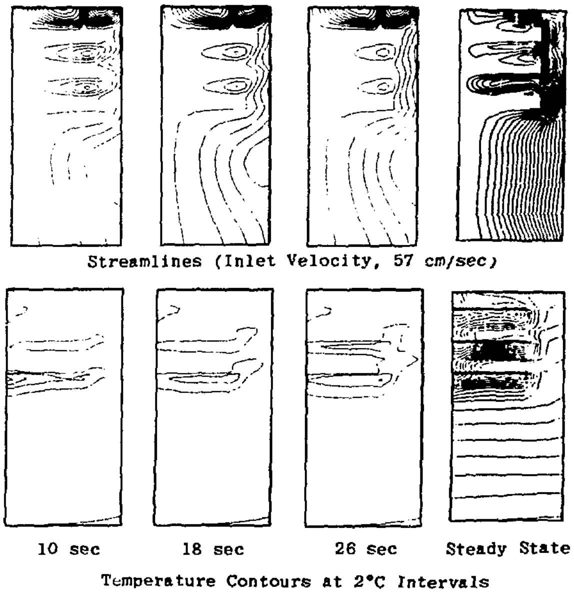

Figure 4b. Continuation of Flow Transient After Pump Shut Down

FINAL OUTCOME

On April 27, 1979, after the calculations were completed, the pump was turned off, natural convection established, and the outlet temperatures monitored. The average outlet temperature rose about 4 degrees then began steadily falling. The predictions and measurement were in substantial agreement in that the outlet temperature did not rise significantly and local boiling did not occur. However sufficiently accurate data on the inlet temperature during the event are unavailable so that a direct comparison between a calculation and measurement is not possible.

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RETRAN NATURAL CIRCULATION ANALYSES DURING THE THREE MILE ISLAND UNIT 2 ACCIDENT

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ABSTRACT

This paper discusses the RETRAN natural circulation analyses which were performed on an emergency basis during the Three Mile Island Unit 2 (TMI-2) Accident in support of the decision to make the transition from forced flow to natural circulation heat removal. The conditions which existed in the containment following the core damage which occurred on March 28, 1979 precluded the long-term use of forced convective core cooling to dissipate fission product decay heat. Natural circulation which uses only the buoyant force of heated water to circulate coolant through the core, was proposed as the most reliable long-term heat removal method. Before the transition could be made, however, it was necessary to determine the capability of the system to successfully achieve stable natural circulation in its damaged state and to evaluate the various modes of steam generator operation which were being proposed for use in this cooling mode.

INTRODUCTION

The Three Mile Island Unit 2 (TMI-2) reactor is designed to make the transition from forced flow to natural circulation core cooling following a trip of the reactor coolant pumps.[1] The final steady state natural circulation core flow is established as a balance between the net elevation head provided by the density changes around the primary loop and the irrecoverable system losses associated with the loop flow rate. The primary irrecoverable losses are geometry and flow dependent. The net elevation head is determined by the elevation of the thermal center of the steam generator relative to the thermal center for the core.

The condition existing at TMI-2 following the accident dictated the employment of natural circulation as the long term cooling mode. Prior to making the transition from forced flow to natural circulation heat removal, it was necessary to determine that adequate core flow could be established for long term cooling despite the increases in system irrecoverable losses caused by the damaged core state. Further, the effect of the mode of operation of the steam generator on the establishment and stability of natural circulation core cooling had to be determined. For example, the effect of using only one steam generator with the other isolated and the effect of the steam generator secondary state, i.e., forced subcooled flow or more conventional steaming saturated conditions, had to be examined.

A detailed two loop RETRAN[2] model of TMI-2 had been previously formulated in 1978 for use in TMI-2 transient and accident studies and had been successfully compared to TMI-2 plant data during the start-up test program[3]. This model was immediately available for use in this application with minimal changes required.

RETRAN MODEL

RETRAN is a state-of-the-art system analysis code developed by Energy Incorporated under the sponsorship and direction of Electric Power Research Institute. RETRAN has been extensively benchmarked by a group of electric utilities for use in a wide variety of system transients[3]. RETRAN solves equations which describe one-component, two-phase compressible flow coupled to heat conduction structures. The application of RETRAN to operational transients is especially accommodated through a flexible scheme used to model interaction of control systems.

A typical configuration of the RETRAN two loop model used for these studies is shown in Figure 1A. As can be seen, this is a very detailed model comprised of 81 volumes, 99 junctions and 31 heat conductors. The detail was more than sufficient to accurately simulate the transition to natural circulation. In particular, the use of 12 axial nodes on both the primary and secondary side of an active once-through steam generator had been previously demonstrated to be adequate to represent the actual primary to secondary heat transfer processes. In many of the cases analyzed in this study, one of the steam generators was isolated on the secondary side and could, therefore, be modelled using a single node secondary as shown in Figure 1A. The pressure vessel includes a seven node core (permitting greater resolution near the top of the core where the damage was presumed to be concentrated), a bypass and a separate downcomer region as well as inlet and outlet plenums. The initial bypass flow fraction could be readily adjusted to evaluate the effect of increased bypass flow resulting from the damaged and possibly reconfigured core region. The representation of four individual cold legs permitted the simulation of reverse flow through three of the loops resulting from operation of one pump prior to the transition. The actual initial flow conditions shown in Figure 1B result from measurements made by B&W following the accident. The asymmetric nature of the initial flow and the steam generator conditions dictated the use of a two loop model for these studies.

Forward direction form losses had been established in previous applications. The reverse form loss coefficients necessary for the idle loops had not been previously established, however. These were initially determined from one pump operation RETRAN analysis using flow splits specified in the TMI-2 Reactor Coolant System Specifications for that plant condition. These were subsequently refined to match the data shown in Figure 1B.

B&W estimated the core pressure drop to be 1.25 x 10^6 Pa. (18.2 psi) for the initial plant condition, i.e. one pump operation with the damaged core. The flow resistence of the upper half of the core was systematically increased in RETRAN until this pressure drop was obtained for the given flow condition. The resistance was increased to approximately 200 times normal indicating the extent of the flow blockage. B&W estimated the bypass flow fraction to be between 20 and 30% of the total flow. Consequently, a 25% bypass flow fraction was used to evaluate its effect on natural circulation.

RESULTS

Coordination with TMI-2 site activities had identified a variety of system states as well as a number of steam generator configurations from which to initiate the approach to natural circulation. A representative set of these initial conditions is shown in Table I. The cases in Table I provide the sensitivity of the system to bypass flow fraction (Cases 1 and 7) and to steam generator secondary condition, i.e. steaming or subcooled flow rate (Cases 7 and 8). In addition, Case 12 closely models the actual transition using initial conditions taken just prior to the pump trip. Case 15, initialized in a natural circulation condition, was used to study the behavior of a switch of one saturated steam generator to the other.

The key system parameters investigated in these analyses were the peak and equilibrium loop ΔT, core flow and the time required to establish stable natural circulation. Knowledge gained in these studies was to be factored into the actual approach to natural circulation. For example, the use of steaming as opposed to forced subcooled flow in the steam generator secondary side could effect the height of the thermal centerline and thus effect natural circulation flow rates. The timing of events and transient behavior was also important in the preparation of contingency plans for problems during transition.

Figure 2A shows the A-loop temperature transient during the transition to natural circulation for the sensitivity cases. All of the cases showed similar trends consisting of an initial peak, several oscillations and finally a reduction to a stable value in the range of 10°C. Figure 2B shows that the final calculated (Case 12) and actual data from the transition are in reasonable agreement witn one another and the sensitivity cases. Figure 2B also shows the A-loop ΔT transients for cases in which an interruption of natural circulation was attempted (Case 15). The figure demonstrates that no such interference could effect the maintenance of long term natural circulation cooling.

Figure 3A shows the core flow rate for the sensitivity cases. A natural circulation flow is established in all cases. In general, core flow reaches a minimum value at the end of the pump coastdown (30 seconds); it then increases as natural circulation begins. The flow oscillates initially but settles down to a steady state after about 2000 seconds. This same effect can be seen for Case 12, that closest to the actual transition, in Figure 3B. Figure 3B also shows the slight effect on flow of the change introduced to the system in Case 15.

CONCLUSIONS

A number of conclusions were drawn from the results of the studies described above including:

i) For the conditions studied, natural circulation is a stable cooling mode despite the effects of the damaged core,

ii) The initial system state prior to the approach to natural circulation did not effect the establishment of stable core cooling,

iii) Once natural circulation was established, it would be difficult to permanently interrupt as long as the primary system remained solid.

In addition, this analysis demonstrates that RETRAN can be effective as a predictive tool for this and similar types of applications and that a large number of cases can be run in a short period of time if a basic plant model is available, benchmarked and properly maintained.

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TABLE I

INITIAL CONDITIONS AND TRANSIENT SUMMARY

Case No.	Core Power, Mwt.	Initial Bypass Flow Fraction (%)	Primary Temperature, °C, (°F)	OTSG "A" Condition	OTSG "B" Condition	Transient
1	5.0	5.4	93.3 (200.)	Forced subcooled flow at 89kg/s (3000 GPM)	Isolated Secondary*	Pump trip and approach to natural circulation
7	5.0	25.0	93.3 (200.)	Forced subcooled flow at 89kg/s (3000 GPM)	Isolated Secondary	Pump trip and approach to natural circulation
8	5.0	25.0	121.1 (250.)	"Steaming" Saturated Secondary*	Isolated Secondary*	Pump trip and approach to natural circulation
12	2.6	25.0	112.8 (235.)	"Steaming" Saturated Secondary* (90% heat removal)	"Steaming" Saturated Secondary* (10% heat removal)	Pump trip and approach to natural circulation
15	2.5	25.0	79.4 (175.)	"Steaming" Saturated Secondary*	Isolated Primary and Secondary*	Switch steam generators after natural circulation is established

*Isolated or steaming secondary of the steam generator had level = 95% of the operating range or 9.6m (31.6 Ft)

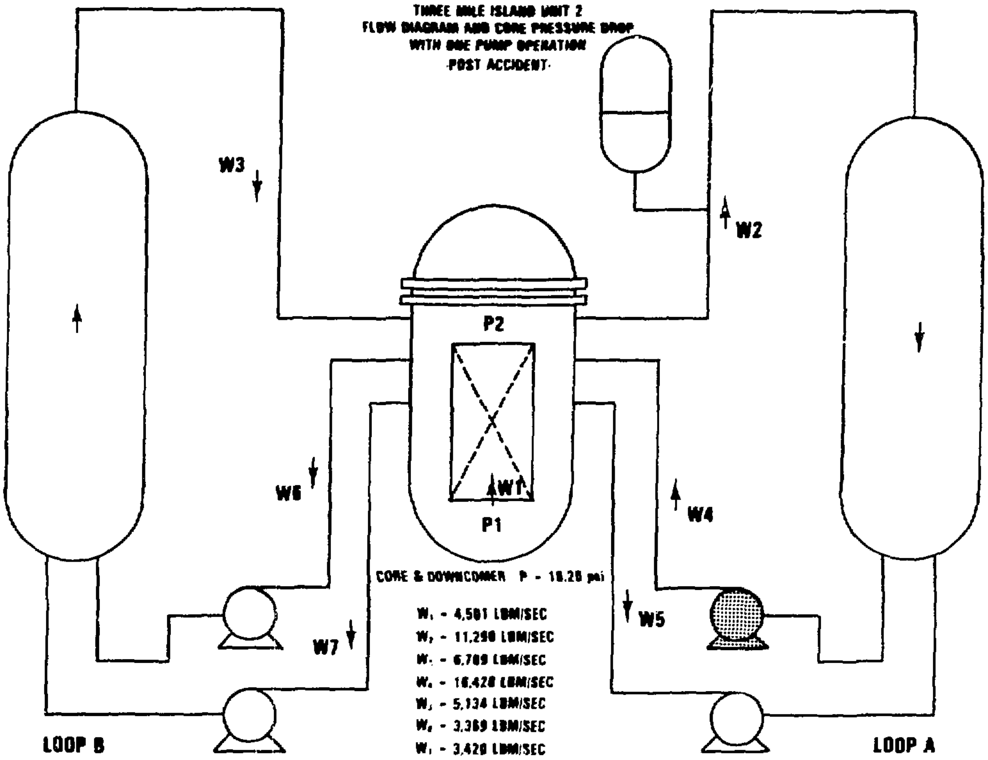

FIGURE 1B

THREE MILE ISLAND UNIT 2
FLOW DIAGRAM AND CORE PRESSURE DROP
WITH ONE PUMP OPERATION
·POST ACCIDENT·

CORE & DOWNCOMER P - 18.28 psi

W_1 - 4,501 LBM/SEC
W_2 - 11,290 LBM/SEC
W_3 - 6,789 LBM/SEC
W_4 - 16,428 LBM/SEC
W_5 - 5,134 LBM/SEC
W_6 - 3,369 LBM/SEC
W_7 - 3,420 LBM/SEC

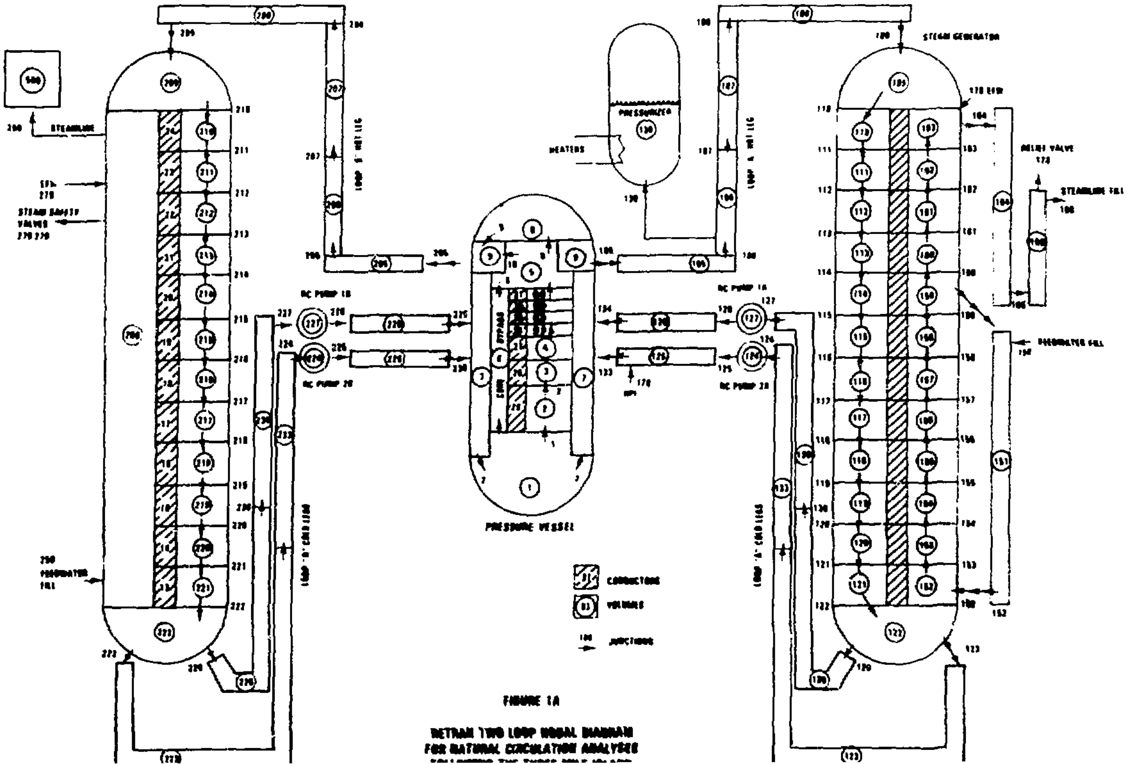

FIGURE 1A

RETRAN TWO LOOP NODAL DIAGRAM
FOR NATURAL CIRCULATION ANALYSES

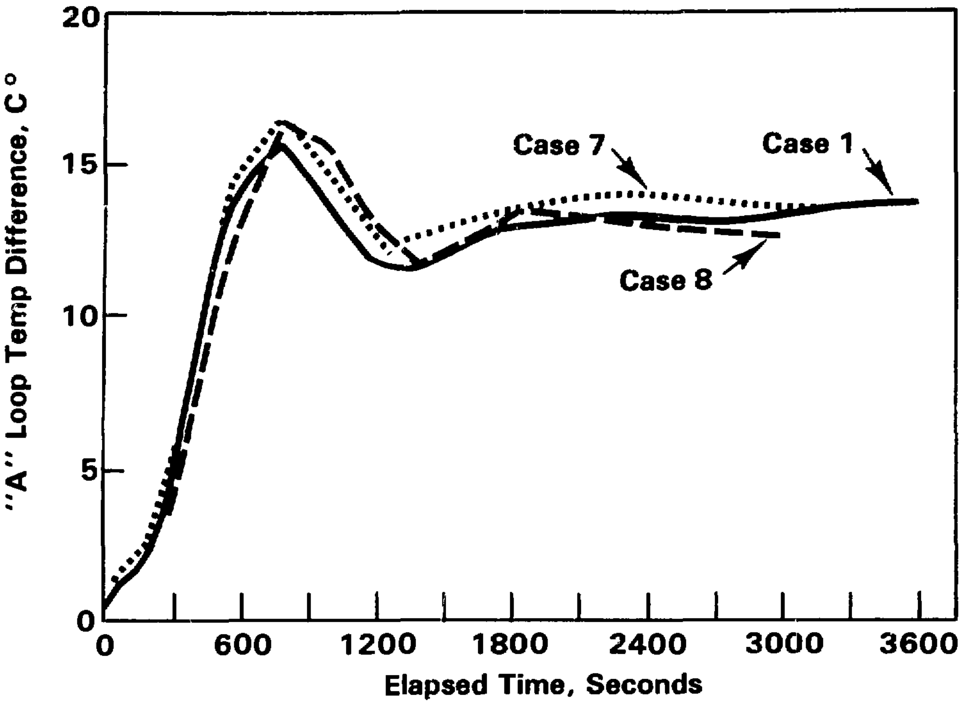

Figure 2A: "A" loop temperature difference for transition to natural circulation (cases 1, 7,8)

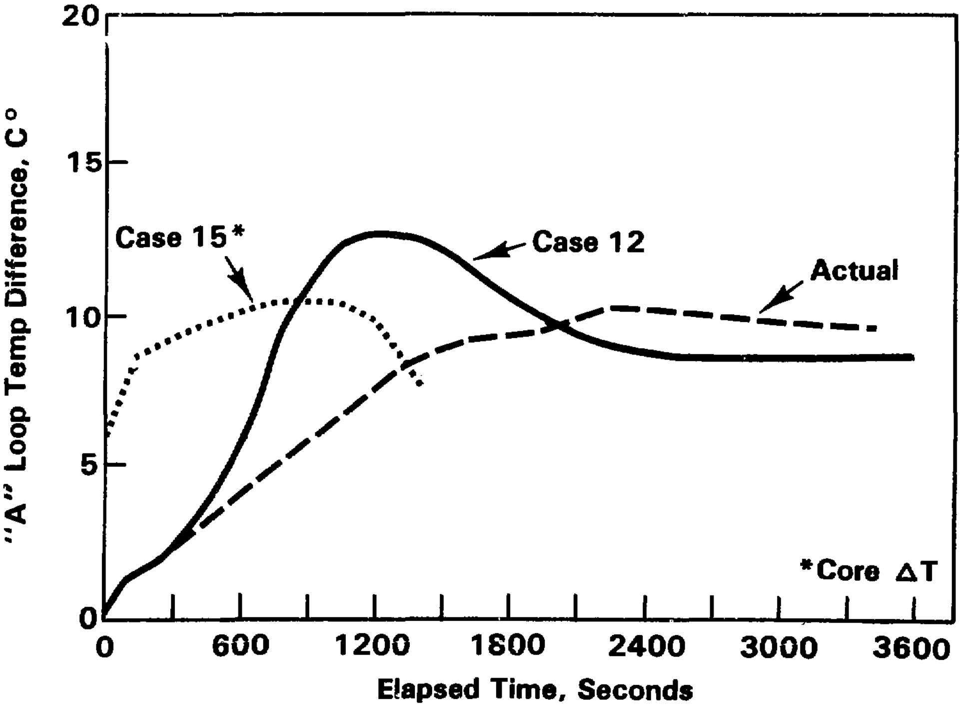

Figure 2B: "A" loop temperature difference for transition to natural circulation (cases 12, 15, actual)

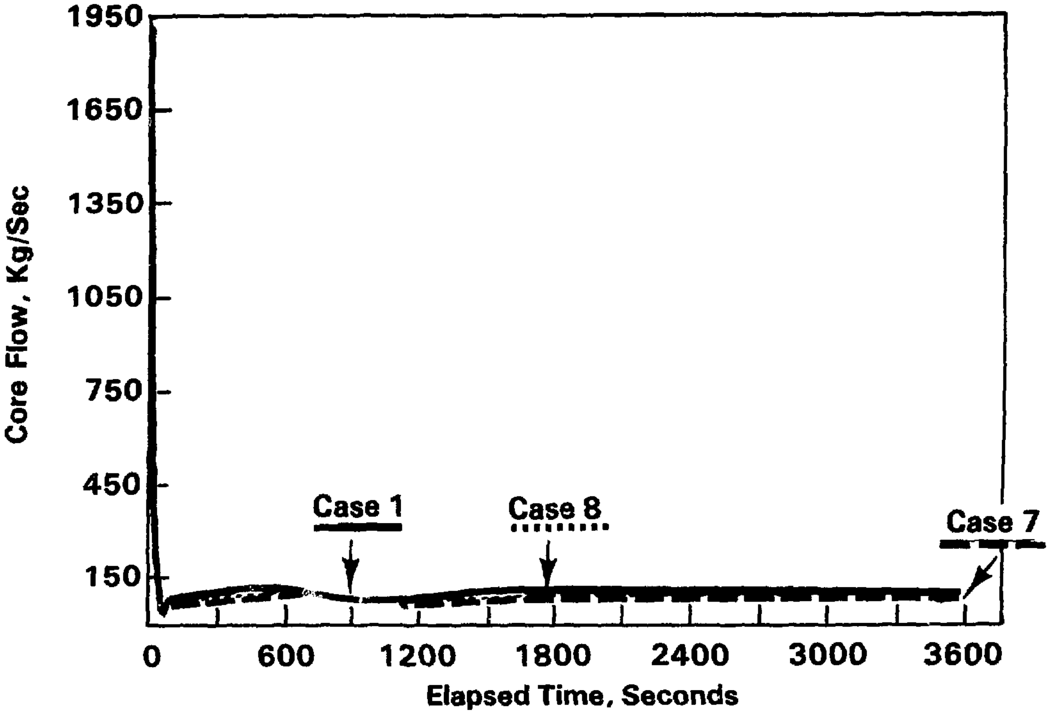

Figure 3A: Predicted core flow versus time for transition to natural circulation (cases 1, 7, 8)

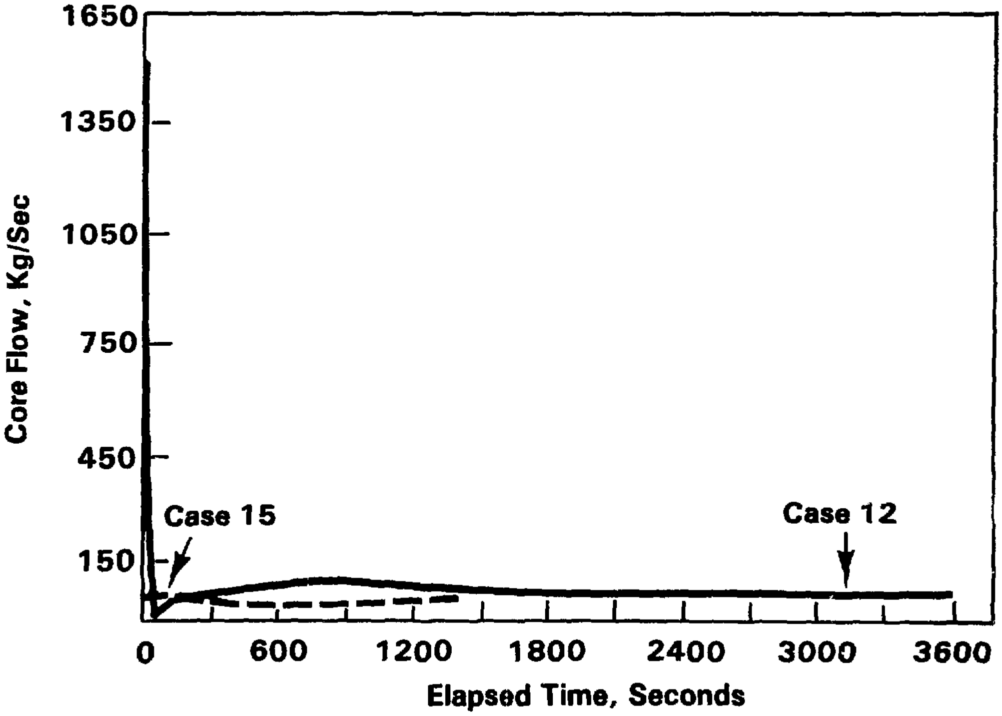

Figure 3B: Predicted core flow versus time for transition to natural circulation (cases 12, 15)

REFLUX BOILING HEAT REMOVAL IN A SCALED TMI-2 SYSTEM TEST FACILITY

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ABSTRACT

An investigation of decay heat removal by the reflux boiling process was performed on a 1/18 linear-scaled test facility simulating the Three Mile Island (TMI-2) primary system. The objective was to clarify reflux boiling phenomena and core cooling effectiveness. Principal test variables included: core power, primary system water and gas inventories, and steam generator secondary-side coolant flow rate. Of 49 tests conducted, 43 achieved a steady-state heat rejection mode within 3 hours. Subsequent analyses identified two distinct reflux boiling modes. Based upon our current understanding, reflux boiling appears to be an effective process for removing decay heat in a broad range of the conditions investigated for a plant of the TMI configuration.

INTRODUCTION

The accident at Three Mile Island, Unit 2 (TMI-2) focused attention on the need for alternate methods to remove decay heat from reactor cores under a variety of emergency conditions [1]. These conditions may include: the unavailability of the normal residual heat removal system, and the presence of steam or condensable gases in the primary coolant system.

Reflux boiling was one of several methods proposed for bringing TMI-2 to a cold shutdown status, particularly if single-phase natural circulation could not be established. In this context, reflux boiling is defined as follows: Primary coolant, entering the reactor vessel from one or more cold legs, is brought to boiling by decay heat from the reactor core. The steam or steam-water mixture thus generated is naturally convected, via the hot legs, to the steam generators in which cooling is provided by circulating water through the secondary side.

The reflux boiling mode differs from the single-phase natural circulation mode in three respects:

- The water inventory in the primary system is less than that required for natural circulation.
- Steam, as well as water, is present in the primary system.
- Noncondensable gases can be present in the primary system.

Reflux boiling technology is well known in industrial applications (e.g., refrigeration cycles), but has not been extensively investigated for potential applications to pressurized water reactors (PWRs) under emergency conditions.

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*Currently on loan to: Yankee Atomic Electric Company, Westborough, Massachusetts 01581.

This investigation clarifies reflux boiling phenomena through tests and analyses of a scale model of a PWR system with once-through steam generators (OTSGs).

EXPERIMENTS

The design and scaling criteria for the test facility were based upon the following considerations:

(1) The primary system configuration was dictated by that of TMI-2, with relevant components and regions represented. Forced convective cooling was provided to the secondary side of the OTSGs. The two cold legs from each OTSG of TMI-2 were combined into one for each loop of the model. Reactor-core decay heat was simulated by electrical heaters in the core region of the reactor vessel model.

(2) The aspect ratio (L/D) and relative volume (V_r/V_{total}) for principal regions within the primary coolant system were maintained approximately equal between the prototype and the model.

(3) Fluid properties (pressures, temperatures, steam quality, and noncondensable gas concentration) were taken on a one-to-one basis between the prototype and the model.

(4) The steam mass flux in the hot leg was set to be the same for the prototype and the model.

(5) From an energy balance on the reactor vessel, combined with Items 2 through 4 above, the decay heat power was scaled as the square of the linear scale factor, approximately 1/18.

Table I summarizes pertinent parameters for the model and the TMI-2 prototype; Figure 1 presents a sketch of the model. The independent and dependent variables for the test system are summarized in Table II.

The evacuated vessel was filled with degassed water prior to testing. Forty-nine experiments, including some reproducibility tests, have been completed. Table II summarizes the range over which the independent test parameters were varied.

ANALYSIS

The analytical model is based upon the concept that steady-state heat transfer from the primary to the secondary side of the active condenser occurs in two vertical regions: single-phase counterflow heat exchange below the primary side water level, and condensation above the water level. When noncondensable gas is present in the primary system, it soon tends to stack above the water level in the condenser. Condensation then occurs immediately above the noncondensable gas layer.

Define Z_1 as the elevation above the lower tube-support plate at which condensation ends and single-phase heat exchange begins, and Z_2 as the total length of the condenser tubes. (Nomenclature is defined in Tables I and II). Local energy balances and boundary conditions are solved for each region to obtain the following profiles:

<u>Single Phase Region ($0 \leq Z \leq Z_1$)</u>--

$$T_s(Z) = T_{s0} + (T_{p0} - T_{s0})[\exp(K_1 Z) - 1]M/(1 - M), \quad (1)$$

$$T_p(Z) = T_{p0} + (T_{p0} - T_{s0})[\exp(K_1 Z) - 1]/(1 - M). \quad (2)$$

Condensation Region $[Z_1 \leq Z \leq Z_2]$--

$$T_p(Z) = T_{sat}, \text{ constant} \quad (3)$$

$$T_s(Z) = T_{sat} - (T_{sat} - T_{s2})\exp[K_2(Z_2 - Z)] \quad (4)$$

$$X(Z) = X_2 - C_p[T_{s2} - T_s(Z)]/Mh_{\ell v} \quad (5)$$

where $M = \dot{m}_p/\dot{m}_s$, $K_1 = N\ D_tU_1(1 - M)/M\dot{m}_sC_p$, and $K_2 = N\ DU_c/\dot{m}_sC_p$. The overall heat transfer coefficients U_1 and U_c [2] are assumed to be independent of Z. To solve the profiles given by Eqs. (1)-(5) and to determine the flow ratio M, we relate the dependent parameters to independent parameters given in Tables I and II. Given our system geometry, primary water inventory (V_w), and hot leg nozzle elevation (Z_{HLN}), the liquid level measurements in the condenser (Z_{SG}) and reactor vessel (Z_{RV}) are correlated by:

$$(Z_{RV}/Z_{HLN}) \simeq (Z_{SG}/Z_{HLN}) \simeq 3.46[(V_w/V_p) - 0.205];$$
$$0.392 \leq (V_w/V_p) \leq 0.474 \quad (6)$$

$$(Z_{RV}/Z_{HLN}) \simeq (Z_{SG}/Z_{HLN}) \simeq 1.22[(V_w/V_p) + 0.289];$$
$$0.474 \leq (V_w/V_p) \leq 0.528 \quad (7)$$

$$(Z_{RV}/Z_{HLN}) \simeq 1; \ (Z_{SG}/Z_{HLN}) \simeq 7.55[(V_w/V_p) - 0.396];$$
$$0.528 \leq (V_w/V_p) \leq 0.718. \quad (8)$$

Since noncondensable gas (not in solution) stacks above the water in the active condenser tubes, then

$$Z_1 = Z_{SG} + 4V_{NCG}/N\pi D_t^2 \quad (9)$$

From a hydrostatic pressure balance between the condenser and hot leg (candy cane elevation is Z_{cc}), the void fraction at the condenser inlet is:

$$\alpha = (Z_{cc} - Z_{SG})/(Z_{cc} - Z_{RV}) \quad (10)$$

Drift flux theory [3] applied to the flow in the hot leg gives the condenser inlet quality as:

$$x_2 = [1 + A_h\rho_\ell v_d/C_0\dot{m}_p]/[1 + \rho_\ell(1 - C_0\alpha)/C_0\alpha\rho_v], \ 0 \leq \alpha \leq \alpha_T \quad (11)$$

$$x_2 = 1.0, \ \alpha_T \leq \alpha \leq 1.0 \quad (12)$$

where

$$\alpha_T \equiv \dot{m}_p/(C_0\dot{m}_p + A_h\rho_v v_d),$$
$$C_0 \equiv 1.2 - 0.2\sqrt{\rho_v/\rho_\ell}, \ v_d \equiv 0.345\sqrt{gD_h(\rho_\ell - \rho_v)/\rho_\ell}.$$

Continuity principles and regional energy balances on the system provide the necessary independent equations for closure:

$$\eta = (\dot{Q}_{in} - \dot{Q}_{loss})/\dot{Q}_{in} \quad (13)$$

$$\dot{Q}_{loss} = \sum_j \pi L_j D_j U_j (T_j - T_\infty) \quad (14)$$

$$T_{s2} = T_{s0} + (\eta\dot{Q}_{in})/(\dot{m}_s C_p) \quad (15)$$

$$T_{p0} + T_{sat} - (T_{s1} - T_{s0})/M \quad (16)$$

$$M = C_p(T_{s2} - T_{s1})/x_2 h_{\ell v}, \quad (17)$$

where T_{s1} and T_{sat} are given by continuity of temperature profiles at Z_1. Finally, the total primary system pressure is obtained by applying Dalton's law of partial pressures, at the saturation temperature, to the gas space.

RESULTS AND CONCLUSIONS

Of the 49 tests conducted, 43 clearly achieved a steady heat-rejection mode. The time to reach steady state, governed by material properties and fluid transport processes, was from ~½ to 3 hours, depending upon the change imposed on the prior state of the system. The power removal capacity of the active OTSGs ranged from 40 to 90 percent of the input power; this increased with secondary-side cooling rate (Figure 2). Heat losses from the insulated pipes and vessels and condensations in the sight-glasses account for the energy balance. Five tests appeared to approach--but may not have reached--an equilibrium state in the time final data were recorded. Only one test (~9% STP concentration of nitrogen gas) clearly did not achieve a steady state within the pressure limit (~100 psi) of the test system.

Two distinct reflux boiling modes within the primary system have been defined from the test data and analyses. Table III summarizes the principal hydraulic process within the primary coolant system and the heat transfer process within the active OTSGs for each mode: The water inventory in the primary system (PSWI) is a significant parameter that distinguishes Mode 1 from Mode 2. Figure 3 shows typical axial temperature profiles on the secondary side of the OTSG that result for each mode. When noncondensable gas was added to the system operating in Mode 1, the excess (that not dissolved in the coolant) rapidly accumulated above the water surface in the primary side of the OTSG and shifted the condensation zone upward (Figure 4). For Mode 2, it is not clear where the excess gas accumulated, but we speculate that it resides in the reactor vessel steam dome, or in the top of the hot leg(s) and the OTSGs or both. The saturation pressure in the primary system was found to be a sensitive function of $\dot{Q}_{in}$ and of the external cooling flow through to the OTSGs (Figures 5, 6 and 7). It also increased with the addition of noncondensable gas to the system for Mode 2 (and for Mode 1 if the added gas volume was sufficient to move the condensation zone temporarily out the top of the OTSG).

An analytical model has been developed for the reflux boiling processes in this type of system. A comparison of the predicted and measured axial temperature profile on the secondary side of the steam generator is shown in Figure 4 for a Mode 1 test.

Based upon our current understanding, reflux boiling appears to be an effective process for removing reactor decay heat from a TMI-like closed nuclear steam supply system with forced convective cooling provided to the secondary side of the OTSGs. The principal limitation of the reflux boiling process appears to be that the steady-state saturation pressure must be less than the pressure limit for the primary system boundary (e.g., safety-relief valve set point).

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2. Bird, R. B., Stewart, W. E., and Lightfoot, E. N., Transport Phenomena, John Wiley and Sons, Inc., New York, p. 286 (1960).
3. Wallis, G. B., One-Dimensional Two-Phase Flow, McGraw Hill Co., New York, pp. 282-294 (1969).

Table I. MODEL PARAMETERS

Parameter	Notation	TMI-2 Value	Model Value
Steam generation	$x\dot{m}_p$	952 gm/sec	2.90 gm/sec
Steam flux in hot leg	$x\dot{m}_p/A_h$	0.146 gm/sec-cm^2	0.146 gm/sec-cm^2
Hot leg ID	d_h	91.4 cm	5.04 cm
Core power	$\dot{Q}_{in}$	2,000 kW	6.2 kW
Primary system volume	V_p	338.7 m^3	0.0569 m^3
Volume of hot leg (one)	V_h	14.9 m^3	0.0025 m^3
Volume of cold leg (one)	V_c	21.1 m^3	0.0035 m^3
Volume of reactor vessel (RV)	V_r	110.6 m^3	0.0185 m^3
$\sum$Tube volume	V_t	38.7 m^3	0.0065 m^3
Length of hot leg	ℓ_h	21.6 m	1.25 m
Length of cold leg	ℓ_c	26.5 m	1.77 m
Heat-exchanger tube ID	d_t	1.41 cm	0.833 cm
Heat-exchanger tube length	ℓ_t	15.8 m	1.46 m
Number of heat-exchanger tubes	N	15,531	78
RPV upper plenum volume	V_{ri}	31.4 m^3	0.0054 m^3
Height of dogleg in cold leg	ℓ_{cd}	1.22 m	0.067 m

Table II. TEST VARIABLES

Variable		Test Value	
Independent	Units	Minimum	Maximum
Input power, $\dot{Q}_{in}$	kW	1.53	9.89
Primary water inventory, V_w	litrs	26.8	44.5
V_w as a percent of V_p	%	47	78
Noncondensable gas vol., V_{NCG}	STP litrs	0	1.96
V_{NCG} as a percent of (V_p-V_w)	%	0	9.3
Secondary coolant flow rate, $\dot{m}_s$	gm/sec	8.3	17.8
Secondary coolant inlet temp., T_{s0}	^{o}C	15	22
Active OTSGs	No.	1	2
Dependent			
Primary system saturation pressure, P_{sat}, and temperature, T_{sat}			
Primary system quality, x, and flow rate, $\dot{m}_p$			
Primary coolant temperature, $T_p(Z)$, in the active OTSG			
Secondary coolant temperature, $T_s(Z)$, along the OTSG			
Energy removal rate by secondary coolant, $\dot{Q}(Z)$			
Heat losses to ambient, $\dot{Q}_{loss}$			

Table III. IDENTIFIED REFLUX BOILING MODES*

(a) Mode 1--PSWI less than amount required to submerge hot leg ports from the reactor vessel.

Mass Flow Rate	Hydraulic Process - Primary System	Active OTSG Heat Transfer Process	
		Primary Side	Secondary Side
$\dot{m}_p/\dot{m}_s < 1$	Liquid levels and static pressures in the OTSG and reactor vessel are about equal; steam exits the hot leg port, flows to the OTSG, condenses in a short axial length of the tubes, and returns via the cold leg to the reactor vessel. Excess noncondensable gas accumulates above the water surface inside the OTSG tubes.	Condensation heat transfer above the liquid (and noncondensable gas) level; single-phase convective heat transfer below this level. Coolant temperature exiting the tubes closely approaches the secondary coolant inlet temperature.	Forced convection heat transfer throughout. Coolant temperature has a sharp gradient in the condensation zone as it absorbs most energy in this region.

(b) Mode 2--PSWI greater than amount required to submerge hot leg ports from reactor vessel.

$\dot{m}_p/\dot{m}_s > 1$	OTSG and reactor vessel liquid levels oscillate mildly. Gas in the upper head depresses the reactor vessel liquid level to the hot leg ports. Water displaced from vessel causes higher levels in the OTSGs. Gas slugs exit hot leg ports, induce slug flow through the hot legs, enhance the coolant flow rate, and result in less steam production than in Mode 1. Steam condenses near the top of the OTSG. Subcooled liquid returns to the reactor vessel.	Condensation heat transfer from the smaller amount of steam in the two phase mixture near the top of the OTSG; single-phase convective heat transfer below the liquid level in the tubes. Coolant exits the tubes with generally less subcooling than for Mode 1.	Forced convective transfer throughout, with generally milder temperature gradients than for Mode 1. Most heat is transferred in the lower half of the OTSGs.
$\dot{m}_p/\dot{m}_s = 1$	Similar to above, but the primary flow rate is nearly equal to the secondary flow rate.	Similar to above, but heat transfer is more or less uniform along the OTSG.	
$\dot{m}_p/\dot{m}_s < 1$	Similar to above, but the primary flow rate is less than the secondary flow rate.	Similar to above, but most heat is transferred in the upper half of the OTSG.	

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*Axial temperature profiles are shown in Figure 3.

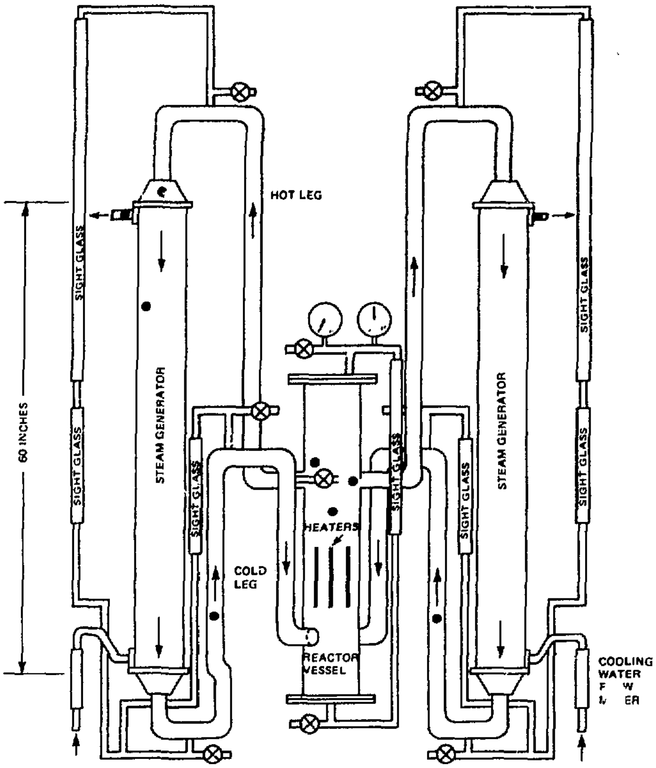

FIGURE 1 REFLUX BOILING MODEL

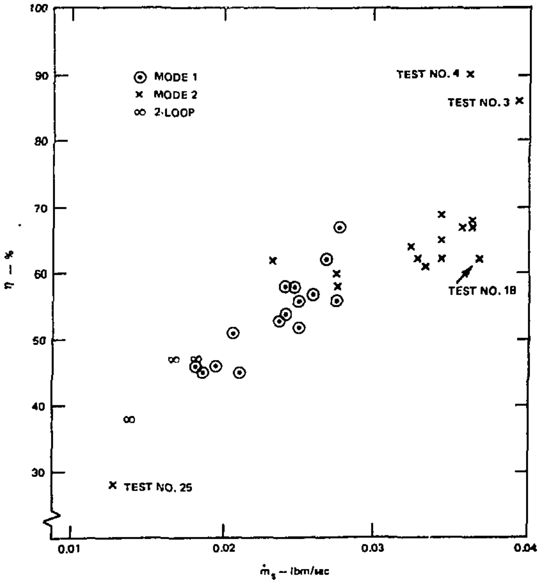

FIGURE 2 EFFICIENCY AS A FUNCTION OF COOLING WATER FLOW

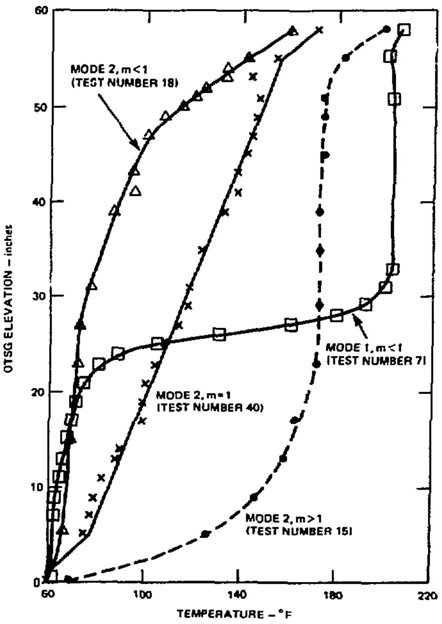

FIGURE 3 AXIAL TEMPERATURE PROFILES OF VARIOUS MODES

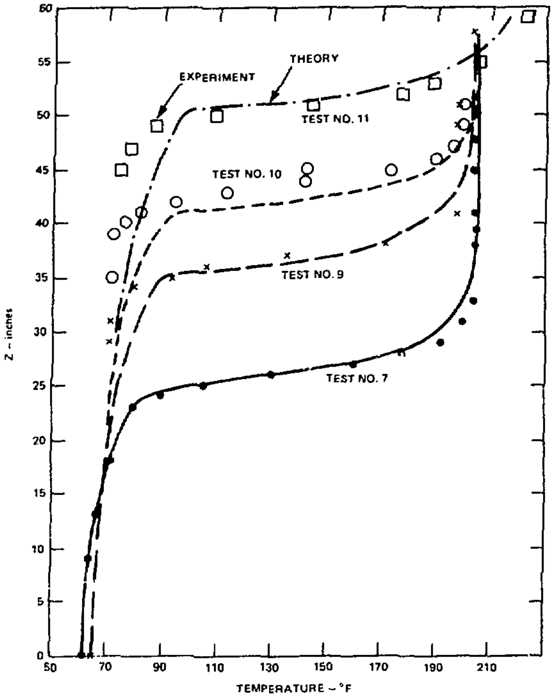

FIGURE 4 EFFECT OF NONCONDENSABLE GAS ON MODE 1

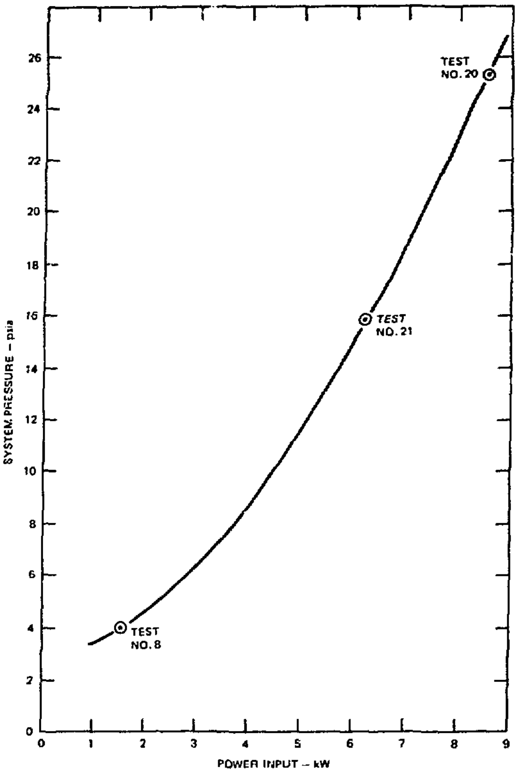

FIGURE 5 PRESSURE AS A FUNCTION OF POWER INPUT – MODE 1

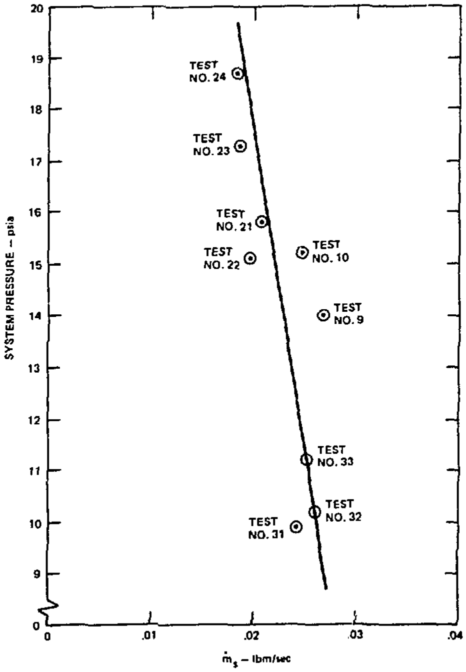

FIGURE 6 PRESSURE AS A FUNCTION OF COOLING WATER FLOW – MODE 1

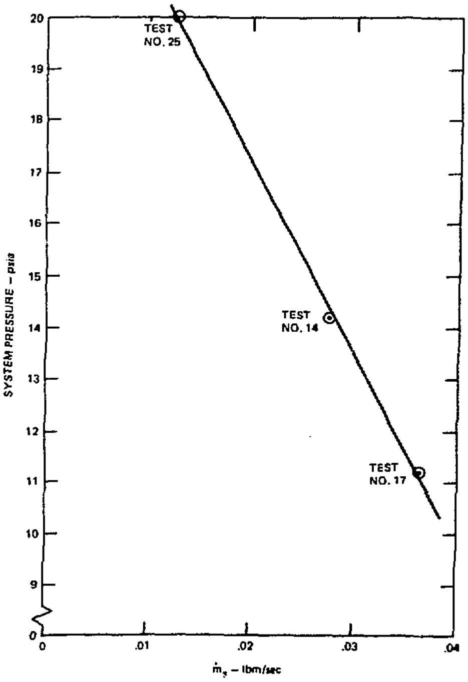

FIGURE 7 PRESSURE AS A FUNCTION OF COOLING WATER FLOW – MODE 2

THE USE OF RETRAN TO EVALUATE
ALTERNATE ACCIDENT SCENARIOS AT TMI-2

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ABSTRACT

At Three Mile Island Unit Two on March 28, 1979, the plant response following a loss of feedwater reflected the combined effects of a delay in initiating emergency feedwater flow, a stuck open primary system relief valve and inadequate high pressure injection flow. The method used to determine the relative effects of each of the separate abnormal conditions was to perform a series of parametric simulations of the accident, using the RETRAN code, in which each of the abnormal conditions was selectively eliminated. This study concluded that the delay in initiation of emergency feedwater flow was not a significant factor and that adequate high pressure injection flow would have provided sufficient core cooling and primary system inventory.

INTRODUCTION

The Three Mile Island Unit Two (TMI-2) reactor experienced a complex transient on March 28, 1979. The manner in which the plant behaved was a function of the transient initiating event, a loss of feedwater, and the subsequent series of malfunctions which followed, including an eight minute delay in the initiation of emergency feedwater, a stuck open primary system relief valve and inadequate high pressure injection (HPI) flow. In order to understand the relative effect which each of these subsequent malfunctions had on the severity of the accident, it was necessary to determine how the accident sequence would have progressed without a given malfunction. Parameter simulations of the accident were performed in which each of the abnormal conditions was selectively eliminated. This paper describes the use of the RETRAN[1] code to perform these analyses and presents the results of three alternate accident scenarios: no delay in emergency feedwater flow, proper primary system relief valve operation and sustained high pressure injection flow.

RETRAN CODE AND TMI-2 MODEL

A computer code and plant specific model which could produce a high fidelity simulation of TMI-2 were necessary to accurately assess the effects of the individual abnormal conditions. Previous analyses using the RETRAN code and the TMI model had shown close agreement with plant data. Previously analyzed events included a TMI-1 Loss of Electrical Load Transient[2], a TMI-2 Stuck Open Steam System Relief Valve Event[3], and TMI-2 Natural Circulation Analyses[4].

An additional benchmark against actual TMI-2 loss of feedwater data was also obtained. This benchmark was useful in identifying appropriate secondary system modeling techniques which were found to be extremely important in the simulation of a transient of this type. The overall approach taken in this study was to first simulate the normal loss of feedwater event, then proceed with a detailed simulation of the accident itself and finally to repeat the accident simulation while excluding selected malfunctions. As indicated above, this approach was taken to provide confidence in our methods and in the results of the sensitivity analyses which are presented herein.

Accident and alternate scenarios were simulated using the two loop model shown in Figure 1. The features of this model which are important to this study include a multi-node reactor vessel head, a turbine bypass valve control system, an emergency feedwater control system and a detailed steam generator model. The multi-node head represents stagnant and low flow regions in which two phase mixtures may result when pressure is rapidly decreased. These regions were found necessary during simulations of other events[3]. The turbine bypass valve control system modulates valve position to control steam system pressure. The emergency feedwater control system modulates flow to control steam generator downcomer level. The steam generator primary to secondary side energy exchange following a loss of feedwater must be modeled with reasonable accuracy in order to obtain an acceptable simulation of primary system behavior. A great deal of effort was thus expended in the modeling of the once thru steam generator. Particular attention was paid to aspirator and downcomer-to-tube-bundle modeling in addition to mass inventories and distributions in the downcomer and tube bundle regions. The purpose of including four nodes per hot leg was to provide greater accuracy in our ability to track system voiding which is likely to occur early in the top of each hot leg. Energy input to the system was from decay heat and pump heat. Ambient heat loss and sensible heat stored in vessel walls and internals were not considered.

For the purposes of comparison, the accident simulation results are included in each alternate scenario figure.

NO EMERGENCY FEEDWATER DELAY

This scenario provides a heat sink but results in a loss of subcooling. The stuck open primary relief valve reduces primary system pressure to the high pressure injection setpoint at about three minutes. The simulation assumes reduced high pressure injection flow beyond four and one half minutes.

Steam generator pressure (Figure 2A) is controlled by turbine bypass valves and emergency feedwater maintains steam generator inventory. In the accident case, lack of emergency feedwater results in a reduction in steam generator pressure and eventual dryout.

With the steam generator as a heat sink, primary system temperature (Figure 2B) is reduced to the steam generator saturation temperature. In the accident case, primary temperature is held constant for four and one half minutes by high pressure injection cooling, then rises when HPI flow is reduced.

In both scenarios, the primary system pressure (Figure 2C) decreases until saturation conditions are reached as a result of the primary relief valve which opens at 8 seconds and remains open during the event.

The primary pressure reduction produces voids in the reactor vessel upper head and primary loops resulting in pressurizer level (Figure 2D) increasing off scale.

PROPER PRIMARY SYSTEM RELIEF VALVE OPERATION

In this scenario no emergency feedwater is available and a loss of heat sink results. The primary system temperature increases to saturation for the primary relief valve setpoint and inventory is gradually lost through the primary system relief valve. System pressure remains above the high pressure injection setpoint so HPI is not available to supply cooling or maintain inventory.

Steam generator pressure (Figure 3A) decreases, as in the accident case, due to dryout of the steam generators.

Primary system temperature (Figure 3B) increases due to decay heat and reactor cooling pump heat. The expansion of primary fluid increases pressurizer level (Figure 3D) and primary system pressure (Figure 3C) to the primary relief valve setpoint. Discharge through the primary relief valve continues to reduce primary inventory. Temperature reaches saturation for the relief valve setpoint and voiding begins in the primary loops at about sixteen (16) minutes.

SUSTAINED HIGH PRESSURE INJECTION

This scenario assumes both a stuck open primary relief valve and no emergency feedwater. Loss of primary subcooling and loss of the steam generator as a heat sink result. High pressure injection is initiated as a result of low primary system pressure and provides an adequate heat sink and maintains sufficient primary inventory.

The first four and one half minutes are identical to the accident case. Steam generator pressure (Figure 4A) decreases due to dryout from lack of feedwater. Primary system pressure (Figure 4C) decreases as a result of the stuck open primary relief valve. This pressure decrease initiates HPI which results in pressurizer level increasing off scale (Figure 4D).

Beyond four and one half minutes, the reduction of HPI flow in the accident case results in a primary temperature increase (Figure 4B). In the scenario in which HPI flow is sustained, primary temperature will gradually decrease as the decay heat rate drops and HPI cooling continues. Primary pressure also continues to drop gradually as a result of HPI cooling.

CONCLUSIONS

This study of three alternate TMI-2 accident scenarios indicates:

i) The eight minute delay in the initiation of emergency feedwater flow had no direct effect on accident severity. Emergency feedwater would not have prevented the pressurizer from filling nor the primary system from reaching saturation conditions. In addition, for this particular event, an adequate heat sink can be provided by high pressure injection.

ii) Proper operation of the primary relief valve will not prevent primary inventory loss during a sustained loss of heat sink.

iii) Sustained high pressure injection flow provides an adequate heat sink and assures adequate primary inventory.

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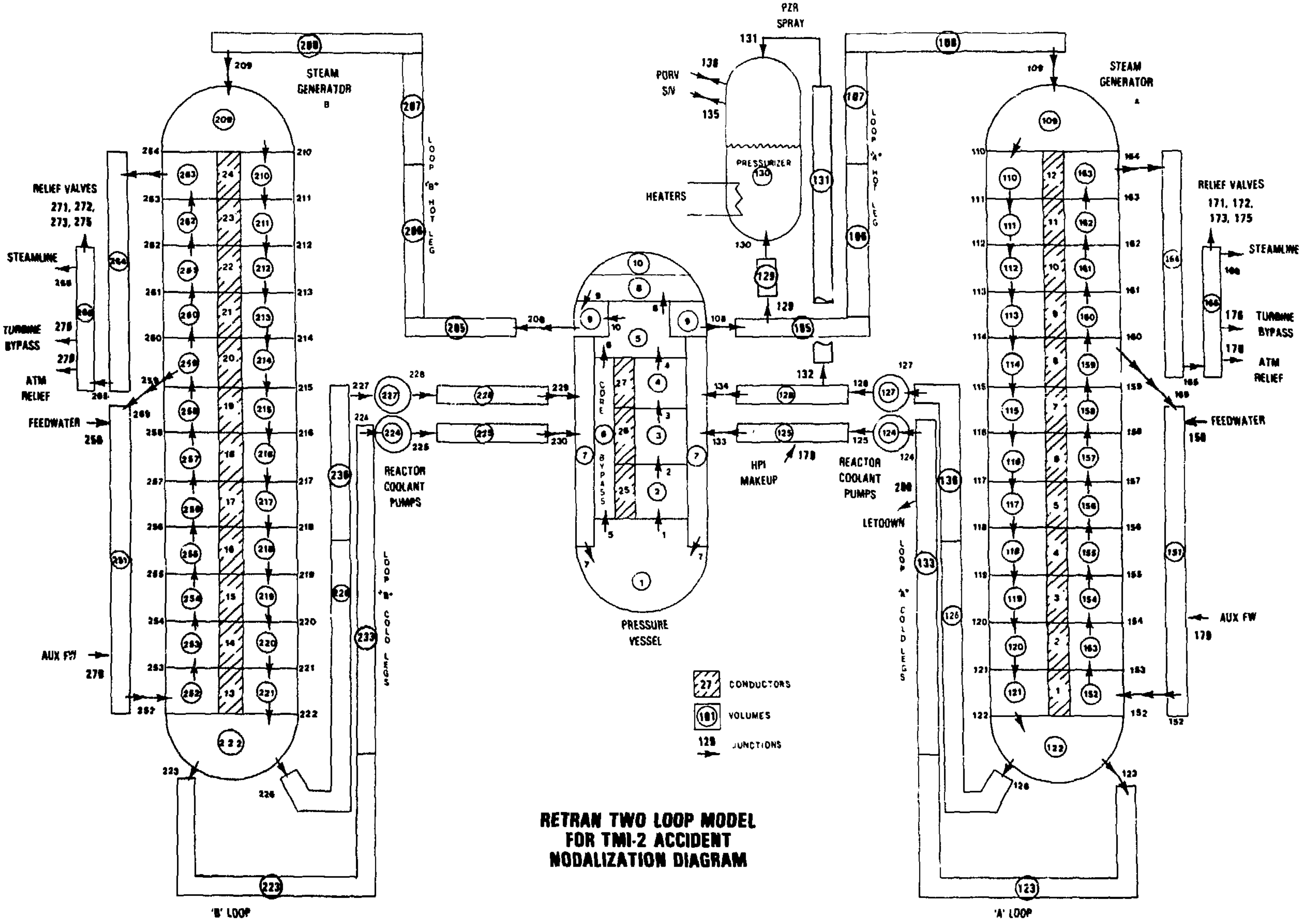
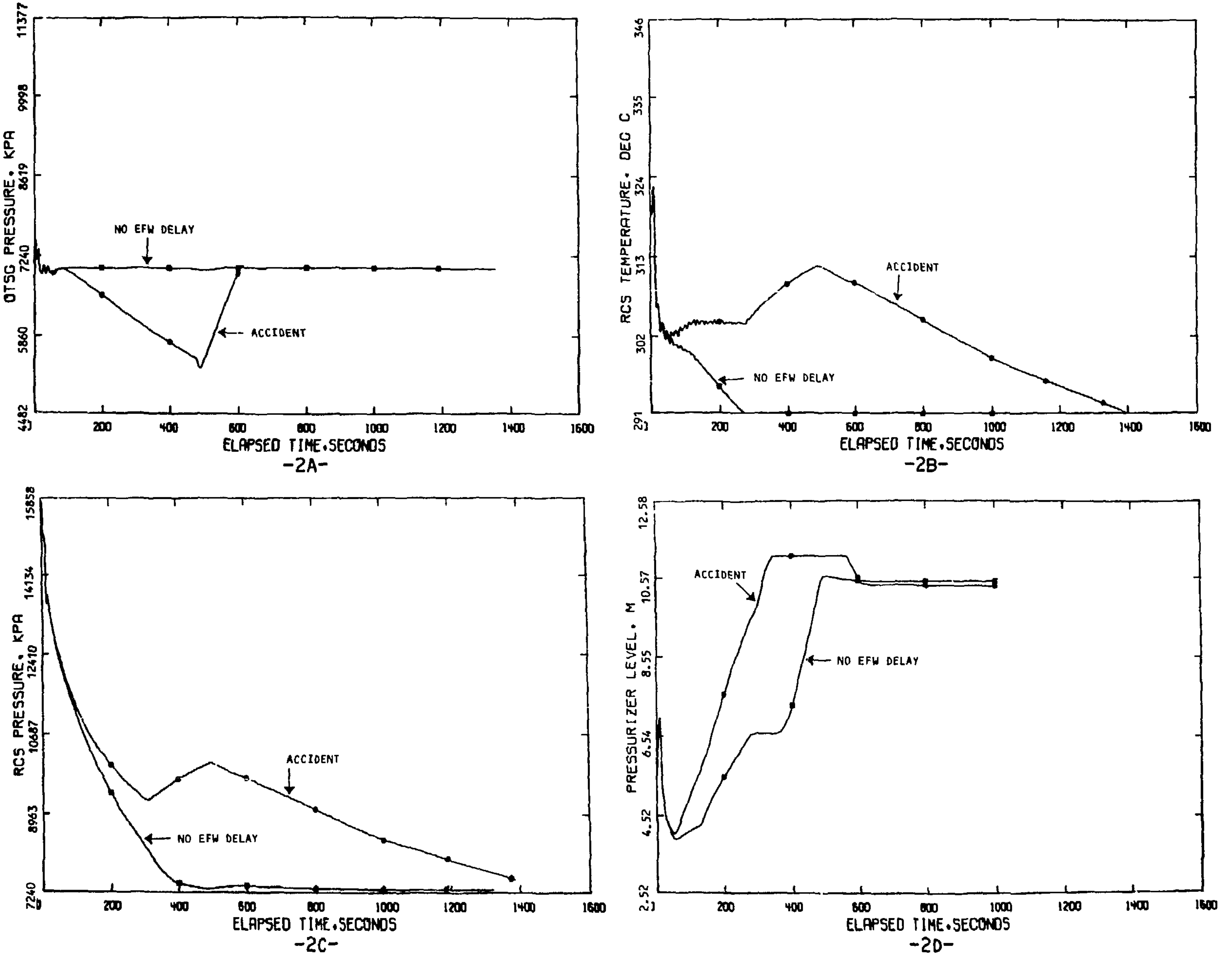
FIGURE 1



RETRAN TWO LOOP MODEL FOR TMI-2 ACCIDENT NODALIZATION DIAGRAM



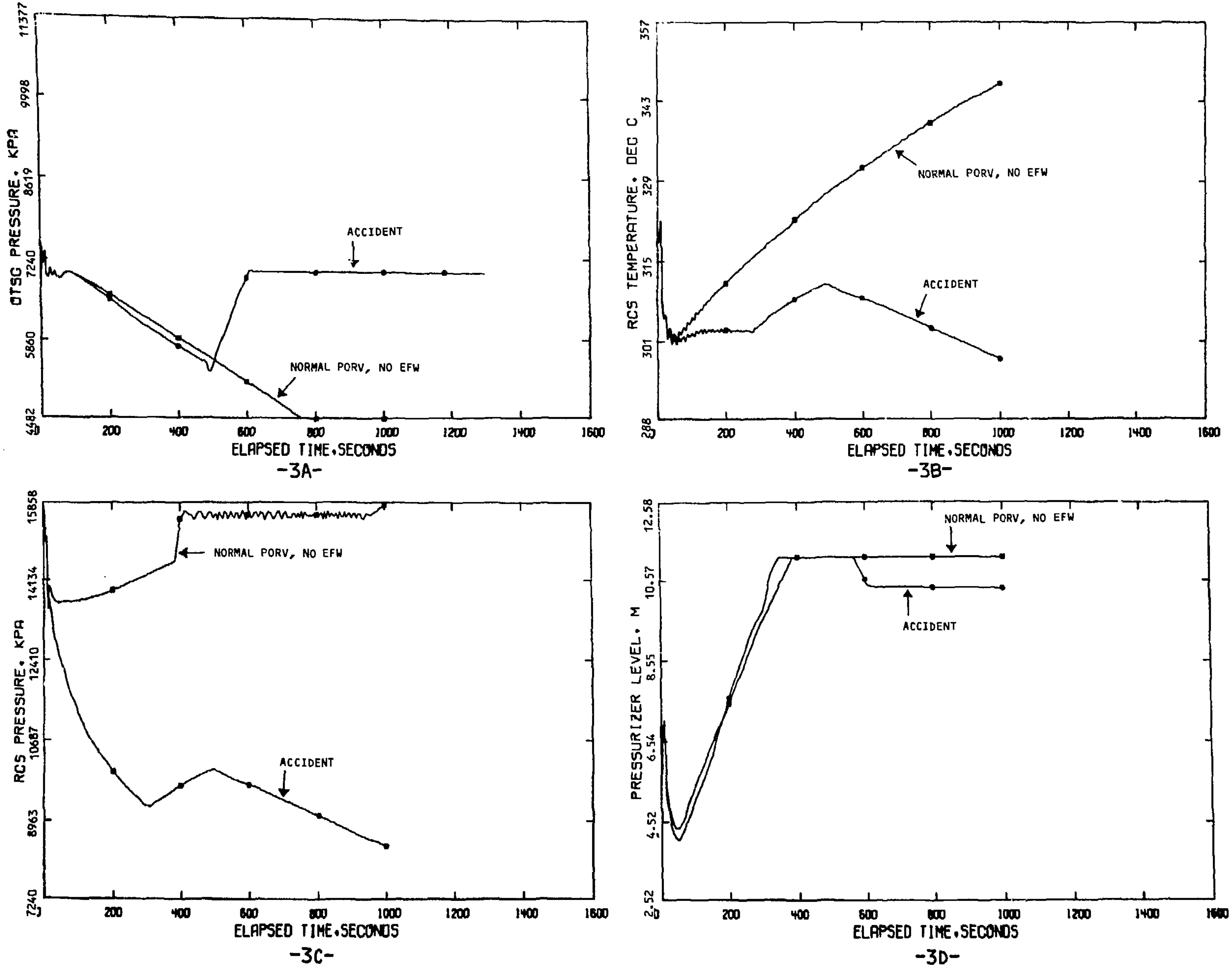
-2A-

-2B-

-2C-

-2D-

FIGURE 2: TMI-2 ACCIDENT SENSITIVITY NO. 1 - NO DELAY IN THE INITIATION OF EMERGENCY FEEDWATER



-3A-

-3B-

-3C-

-3D-

FIGURE 3: TMI-2 ACCIDENT SENSITIVITY NO. 2 - NORMAL PORV WITH NO EMERGENCY FEEDWATER

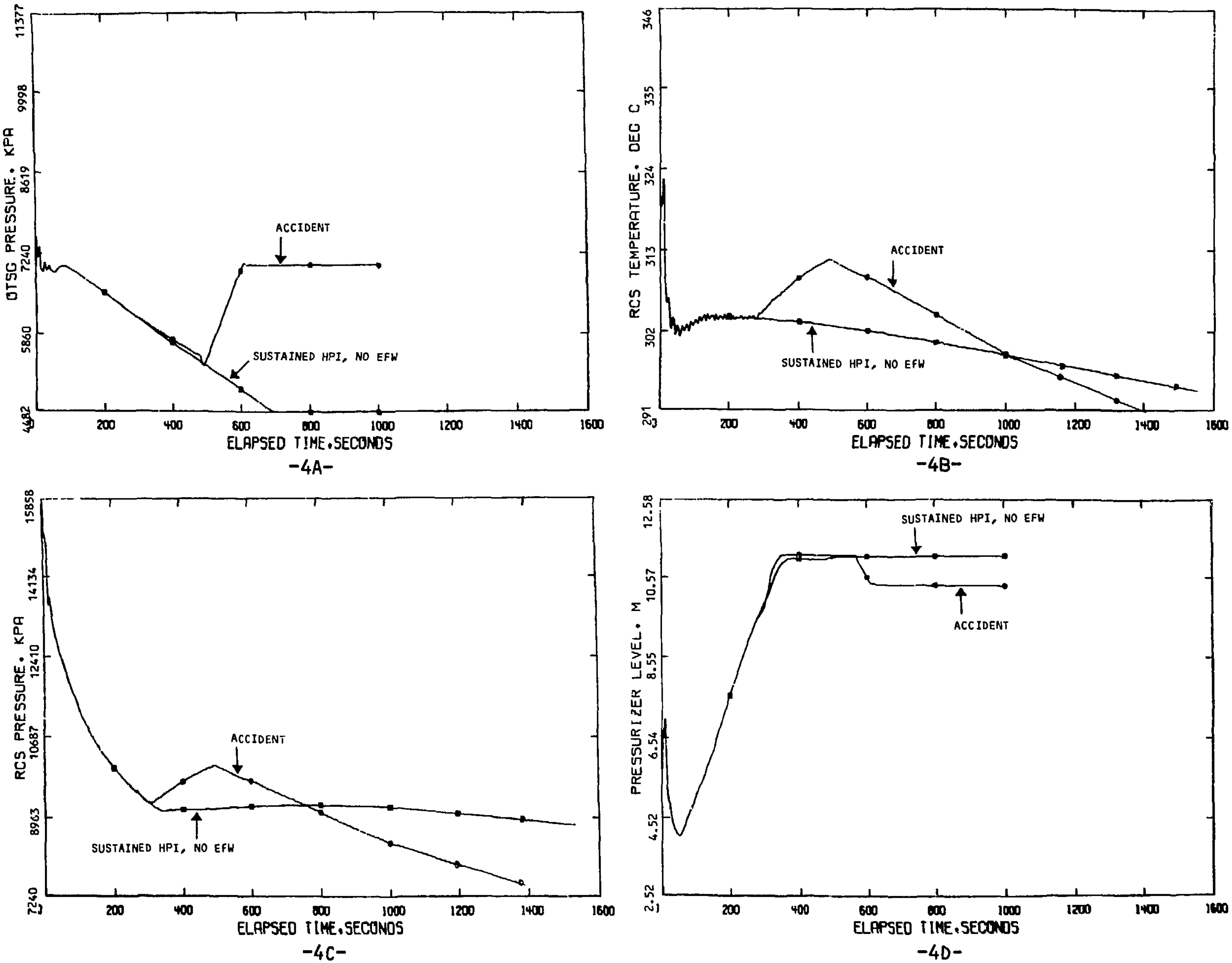


FIGURE 4: TMI-2 ACCIDENT SENSITIVITY NO. 3 - SUSTAINED HIGH PRESSURE INJECTION WITHOUT EMERGENCY FEEDWATER

932

SESSION XVI

A. STATISTICAL ASSESSMENT OF POTENTIAL ACCIDENTS

B. VERIFICATION OF COMPUTATIONAL METHODS

Chairmen

P. H. Govaerts - Association Vincotte

W. T. Russell - U. S. Nuclear Regulatory Commission

ON THE USE OF BAYES' THEOREM IN ASSESSING THE FREQUENCY OF ANTICIPATED TRANSIENTS

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ABSTRACT

A series of examples is given illustrating the use of Bayes' theorem to express our state of knowledge about the frequency of reactor transient events. The examples address the effect of the prior, the mesh spacing, and the interpretation of the data. Application is suggested to the question of completeness, i.e., to transients not yet identified.

PURPOSE

The purpose of this paper is to illustrate by example the use of Bayes' theorem in predicting the frequency of rare, or infrequent events. The specific examples chosen are some of the Transient Events defined in the EPRI publication, EPRI NP-801 [1]. The historical data relevant to those events is also collected in that document.

We shall show how one may use Bayes' theorem to summarize the implications of the historical data; what can and cannot be inferred from such data with what degree of confidence. As one of the examples, we shall illustrate what can be said about events which have not yet occurred, and as a particular case of that, shall address the question of completeness, i.e., of events not yet thought of.

GENERAL COMMENTS ON BAYES' THEOREM

Bayes' theorem and its use has been the subject of controversy ever since it was introduced.[2] It has been called "unscientific" and "unnecessary."[3] It has been accused of "giving rise to inconsistencies" and of being unusable and difficult to get a grip on.[3, 4] On the other hand, it has had sturdy advocates and numerous applications in the nuclear field.[5-8]

The authors do not wish here to enter into a general philosophical argument of the issue. We shall simply state our position and then let our examples speak for us further. Our position is that the above criticisms are incorrect and misleading; that they arise out of misunderstanding what Bayes' theorem is; that far from being unscientific, Bayes' theorem is in fact the fundamental law of logical inference, of reasoning from limited data to general conclusions, and that far from being unncessary, it is not only the best way to deal with such problems, it in fact cannot be avoided; it can only be hidden or obscured, and even then only at the cost of great contortions.

EXAMPLE 1 - BWR EVENT #9 PRESSURE REGULATOR FAILS OPEN BETWEEN 25 AND 100 PERCENT POWER

The data for this example is taken from Table C-86, page C-45 of reference [1]. We observe in that table that there has been a total of 9 such events in 43 reactor years of operation. This is our experience B. We ask what can be inferred from this evidence about the underlying frequency of this type event.

THE MODEL

Before applying Bayes' theorem to this question we note that in asking for the underlying frequency we have already presumed a certain model for the observed phenomenon. We assume, with this question, that there is an underlying frequency, λ, of this event, and that the probability that a reactor will experience this event between time t and t+dt is:

$$p(\text{event \#9 occurring between } t \text{ and } t+dt) = \lambda dt \qquad (1)$$

We assume, moreover, as part of this model that all the reactors obey this same equation and that λ is independent of time, so that each reactor is "good as new" at any instant. According to this model, the well-known Poisson process, the probability that we would experience precisely n events in T reactor years is:

$$p_n(T) = \int_0^T p_{n-1}(\tau)\ p_0(T-\tau)\lambda d\tau = \frac{(\lambda T)^n}{n!}\ e^{-\lambda T}. \qquad (2)$$

Having established these properties of our model we may now proceed to the application of Bayes' theorem.

APPLICATION OF BAYES' THEOREM

For purposes of numerical calculation let us now discretize the λ axis and write Bayes' theorem in the form:

$$p(\lambda_i|B) = \frac{p(\lambda_i)p(B|\lambda_i)}{\sum_{\lambda_i} p(\lambda_i)p(B|\lambda_i)} \qquad (3)$$

where $p(\lambda_i\|B)$	probability of frequency λ_i, given information B.
$p(\lambda_i)$	probability of frequency λ_i, prior to having information B.
$p(B)$	prior probability of B.
$p(B\|\lambda_i)$	probability of B, given that the frequency is λ_i.

Now B is of the form:

$$B = \{n \text{ events in } T \text{ years}\} \tag{4}$$

Therefore, from (2)

$$p(B|\lambda_i) = \frac{(\lambda_i T)^n}{n!} e^{-\lambda_i T}. \tag{5}$$

It remains only to specify the prior distribution $p(\lambda_i)$. For this purpose let us imagine that our initial state of knowledge, prior to the operating experience B, is expressed in the histogram:

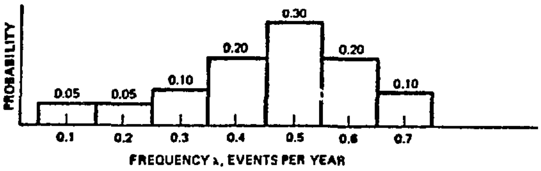

FIGURE 1. PRIOR DPD

With this prior, and with (5) we carry out the calculations of (3) as follows:

TABLE I. BAYES' THEOREM CALCULATIONS

λ_i	.1	.2	.3	.4	.5	.6	.7	Σ
$p(\lambda_i)$	.05	.05	.10	.20	.30	.20	.10	1.0
$p(B\|\lambda_j)$	.0188	.1305	.0681	.0123	.0012	.000087	~0	
$p(\lambda_i)\ p(B\|\lambda_i)$	.00094	.00653	.00681	.00246	.00037	.00002	~0	.01713
$p(\lambda_i\|B)$	.055	.381	.398	.144	.022	.001	~0	1.000

The last row here is our posterior probability distribution after having the information B. It is interesting to compare the posterior and prior graphically, as in Figure 2. We see then that the evidence is sufficient to give us high confidence that the frequency f will not be in the .5 to .7 range, but rather in the .1 to .4 range.

SENSITIVITY TO CHOICE OF PRIOR

One of the questions that inevitably bothers people in connection with Bayes' theorem is, "What if you had chosen a different prior?" It turns out that the choice of prior is not all that important. As long as the choice is within reason only small differences in posterior will result and these will be exactly in accord with what one would expect by common sense. In any case, the story told by Bayes' theorem is expressed mainly in the change to the prior wrought by the evidence B, not by the prior or posterior itself.

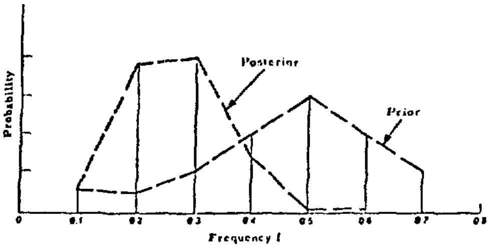

FIGURE 2. PRIOR AND POSTERIOR
FOR BWRs - EVENT #9

To illustrate these facts we repeat the previous calculation using a different prior--in this case, a uniform distribution between .1 and 1.0--indicating an initial state of near complete ignorance. The results are compared in Figure 3 with the posterior from the previous section. As can be seen, there is on the whole relatively little sensitivity. The extension of the prior to the right has no impact since λ values in the range of .7 or greater are totally incompatible with the observed evidence. The granting of greater prior credibility to the values 0.1 and 0.2 does shift the posterior somewhat towards these values exactly as one would expect.

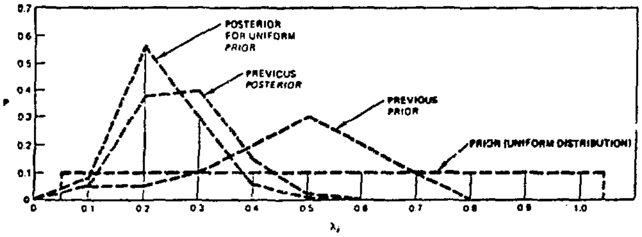

FIGURE 3. EFFECT OF CHOICE OF PRIOR

EXAMPLE 2

We next give an example to show how Bayes' theorem signals us when the prior is inconsistent with the data. Consider Event #18, Abnormal Startup of Idle Recirculation Pump [1].

Suppose we were given the prior DPD shown in Figure 4. From EPRI NP-801, page C-12, the experience is zero events in 49 reactor years. Combining this evidence with the prior yields the posterior shown.

The fact that the posterior DPD does not come down on the left side is a signal to us that our prior did not go far enough on the left. In fact, looking at it again the whole prior is confined to the narrow range .1 to .2. On reflection we realize that we had no reason to confine it this closely--our state of knowledge was nowhere near that "tight."

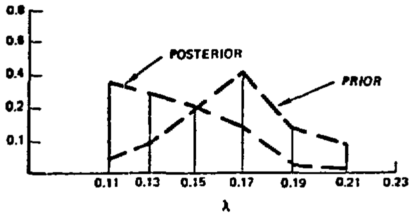

FIGURE 4. COMPARISON OF POSTERIOR AND PRIOR DPD

Therefore, we repeat the calculations with a wider, Alternate Prior histogram which leads to the following result:

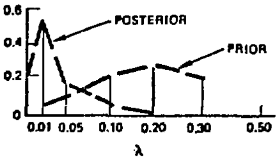

FIGURE 5. RESULTS WITH WIDER PRIOR

In words, Bayes' theorem is telling us that the experience of zero events in 49 years is pretty strong evidence that the actual frequency is very small, in the range of .01 or less. In light of this evidence it is highly unlikely that the ultimate frequency will turn out to be as high as .1 per year or greater. Incidentally, note in this example that the fact that the event in question has never occurred creates no problem in the application of Bayes' theorem. Zero occurrences is just as valid a piece of evidence as any other number of occurrences and is treated in exactly the same way with Bayes' theorem.

EXAMPLE 3 - BWR #13 TURBINE BYPASS OR CONTROL VALVES CAUSE INCREASED PRESSURE (CLOSED)

The purpose of this example is to illustrate the effect of the discretization and to suggest a way of handling the question of whether to include in the calculations the experience of the first plant operating year.

INITIAL STATE OF KNOWLEDGE

Let us assume that we are very ignorant about this event, and reflect this in a broad prior:

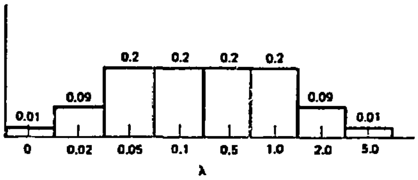

FIGURE 6. PRIOR

From EPRI NP-801, page C-9, we have 25 events in 49 plant years. However, 17 of the events occurred in the first year. Omitting the first year we have 8 events in 37 years. Let us define therefore two sets of information: $B_0 \equiv 25/49$; $B_1 \equiv 8/37$; and do the calculations with both as follows:

TABLE II. BAYES' THEOREM CALCULATIONS

λ	0	.02	.05	.1	.5	1.0	2.0	5.0	Σ
$p(\lambda)$	.01	.09	.20	.20	.20	.20	.09	.01	
$p(B_0\|\lambda)$	0	$\sim 10^{-25}$	$\sim 10^{-16}$	$\sim 10^{-10}$	.0791	.000061	10^{-18}	0	
$p(\lambda)\ p(B_0\|\lambda)$	0	~0	~0	~0	.0158	.00012	~0	0	.016
$p(\lambda\|B_0)$	0	~0	~0	~0	.99	.01	~0	0	
$p(B_1\|\lambda)$	0	1.1×10^{-6}	.000535	.0215	.00314	1×10^{-8}	~0	~0	
$p(\lambda)\ p(B_1\|\lambda)$	0	2.0×10^{-8}	.000107	.00431	.000629	~0	~0	~0	
$p(\lambda\|B_1)$	0	~0	.021	.84	.14	~0	~0	~0	

What Bayes' theorem is telling us now is that 25 out of 49 or 8 out of 37 is a fairly good statistical sample and that the original broad prior is thus greatly narrowed by the evidence, in the first case to the neighborhood of $\lambda = .5$, and in the second, to the neighborhood of $\lambda = .1$. We may therefore, if we wish, revise our discretization using a finer grid in the neighborhood of .1 and .5. For example, we might choose the grid and the uniform prior shown in Figure 7.

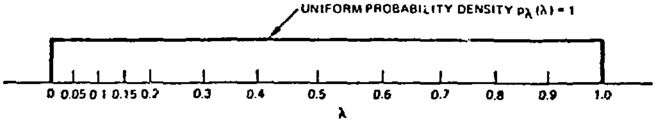

FIGURE 7. UNIFORM PROBABILITY DENSITY ON NONUNIFORM GRID

There remains finally the question of whether B_0 or B_1 is more appropriate. The answer is obviously somewhere in between, a composite, as sketched in Figure 8. Exactly where to place this composite is basically a matter of judgment, and that is where we shall leave it for the time being.

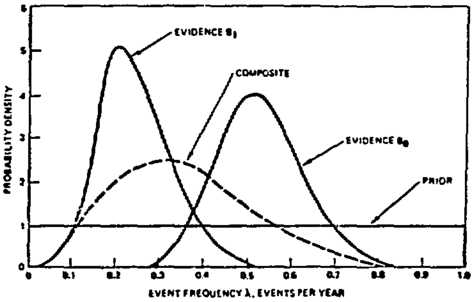

FIGURE 8. BWR EVENT #13

THE ARGUMENT ABOUT COMPLETENESS

An analysis of reactor safety, or indeed any risk analysis of any type, must boil down ultimately to a list; a list of envisioned accident scenarios, together with an assignment of frequency and consequence to each. Given that this is fundamentally the meaning of a risk analysis, and given that in practical cases the list must be finite, then it is easy for a critic to say, "I refuse to put any credibility in your risk analysis because your scenario list is finite, therefore incomplete. What about the scenarios you haven't thought of yet?" Such a critic boasts that he could never be satisfied by any analysis no matter how thorough, and thus justifies implacable opposition to the action which is being analyzed for risk.

We can put this position in perspective by simply pointing out that every decision has at least two branches. Thus, if one is going to be eternally dissatisfied with the risk analysis of one branch, he must also be dissatisfied with the risk analysis of the other branch. Too often the fundamental error is made of criticizing the risks present in one branch without even thinking about the risks of the other branch, much less attempting anything like a definitive listing of scenarios, probabilities, and consequences.

Nevertheless, we should like here to point out another line of argument. Let us imagine that after we have made our list of identified scenarios, we include one more--call it the unidentified scenario. This scenario represents all the events we have not yet thought of, or not otherwise included in the list. With this scenario included we can now logically assert that our list is complete. All possible events are included either as an identified scenario or in the catch-all category.

We can now assign at least an upper bound measure of damage to events in this category from our knowledge of the physics, the maximum energy or radioactivity released, etc. It remains, however, to assign a frequency to this category. How can we assign a frequency to events we have not even defined?

This seems an impossible assignment. However, within the Bayesian framework we can give an answer along the following lines: Let c denote the catch-all category. We seek to estimate the frequency, λ_c, of events of this category. No events of this category have yet occurred, for if they had we would have included them as an identified scenario. Therefore, the evidence, B, for this category is: zero events in T years, where T is however many operating years of experience we have. We may now apply the Bayesian process with, say, a uniform prior to obtain a p versus λ curve for this event as for any other.

CONCLUSION

We have illustrated by example how Bayes' theorem can be used to express our state of knowlege of the frequency of rare events based on limited information. In the various examples we have illustrated the effect of the prior and the choice of discrete mesh. Finally, we pointed out that the bugaboo of "events not yet imagined" can also be addressed from this viewpoint; that information about the frequency of such events is contained in the fact that none of them have occurred yet; that this information can be quantified as for any other event. Thus, we get a measure of the size of the bugaboo and thereby reduce his fearsomeness. Thus, in the face of uncertainty, we can take optimal decisions and act to our benefit, rather than wallow in fear, paralysis, and inaction--which is invariably the very worst branch of the decision tree.

ACKNOWLEDGEMENT

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APPLICATION OF ADVANCED UNAVAILABILITY MODELS TO REDUNDANT SAFETY SYSTEMS

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ABSTRACT

Comprehensive availability models and computer codes have been developed for analyzing redundant standby safety systems with components tested periodically. Common-mode failures, failures not detectable by periodic tests and a variety of other human and hardware failures have been included in the models for consecutive, staggered and random testing schemes. Using model parameters extracted from recorded data, the methods have been used for sensitivity studies and for optimizing the redundancy and testing. Applications to an auxiliary feedwater system of a typical PWR 3-loop design indicate that diesel generators with high failure rates are critical components, but common-mode failures currently limit the benefits of optimization and increasing redundancy.

INTRODUCTION

Analytical models have been developed recently for evaluating the unavailability of a periodically tested component and a train of components in series [1], as well as for safety systems with m-out-of-n (m/n) redundancy [2]. The expressions take into account contributions made by testing, repair, equipment failure, human error and different testing schemes. In this paper common-mode failures and failures not detectable by periodic tests are included and the models are applied to an auxiliary feedwater system (AFWS) of a typical 3-loop pressurized water reactor (PWR) plant. Optimum test intervals and testing strategies as well as the importance of various failure modes are analyzed in detail. Suggestions for design changes can be derived from the results.

The models used in this paper combine and extend earlier analytical works [3-5]. One of the goals is to obtain comprehensive yet simple analytical expressions suitable for design and optimization studies. Computer codes have also been developed to perform the calculations.

AVERAGE UNAVAILABILITY

We assume that the components of a safety system are tested periodically with a test interval η. Each of the test intervals begins with a test of duration υ, followed by a repair period of duration τ (if repair is needed), after which the component remains on standby for the remaining time $\eta-\upsilon-\tau$. A component is assumed to be out of service for a fraction q_0 of the testing time υ.

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The following parameters are taken into account for individual components: λ = failure rate of detectable failures, ρ = probability of failure due to a true demand, γ = probability of failure due to a test, with the component not repaired before the next test, p = probability of failure due to a test, with the component repaired after the test and ω = probability that a failed component does not get repaired during the repair period.

The pointwise unavailability U(t) of a component can be defined as the probability of failing to perform upon demand that occurs at time t. In case of periodic testing this function resembles the sawtooth-and-step-curve used in the FRANTIC code [6], except that the contributions of γ, ρ and ω are explicitly included. When true demands occur at random times with a frequency that is small compared to the testing frequency, the unavailability per demand is equal to the average unavailability over a test interval [1]. In practical cases this can be written as

$$\bar{U} = \rho + \gamma + \lambda\tau + \frac{q_0\upsilon + (p + \gamma)\tau}{\eta} + \frac{\lambda\eta}{2} \quad , \tag{1}$$

when each term individually is small compared to unity ($\bar{U} < 0.1$). The effect of ω is small, and it can be taken into account by increasing the effective repair time from τ to $\tau + \omega\eta$. The minimum of $\bar{U}$ is obtained with the test interval

$$\eta_0 = \left\{ 2[q_0\upsilon + (p + \gamma)\tau]/\lambda \right\}^{1/2} \quad . \tag{2}$$

For a general m-out-of-n redundant system, with n components in parallel and only m needed to perform the safety function, the average unavailability can be written in a polynomial form

$$\bar{U}_{m/n} = \frac{\alpha}{\eta} + \sum_{i=0}^{n} \beta_i \eta^i \quad , \tag{3}$$

when at any time not more than one component is undergoing testing. The constants α, β_i are independent of η but depend on the testing scheme. In a uniformly staggered scheme, the testing of the components is performed at times 0, η/n, $2\eta/n, \ldots, (n - 1)\eta/n$, relative to the first test. In a nearly simultaneous testing scheme, component 2 is tested immediately after the test of component 1 is completed, component 3 is tested after component 2 is tested etc., for each of n components. For systems with $1 \leq m \leq n \leq 4$ the coefficients are given in Ref. [7] for uniformly staggered and nearly simultaneous testing schemes. The ICARUS code has been programmed to calculate the average unavailabilities, optimum test intervals and contributions from testing, repair and other failure modes.

Simple approximate equations can be derived by assuming testing to take place at randomly shifted times within the average interval η. This is equivalent to assuming the average unavailability of Eq. (1) to be valid throughout the interval. This yields

$$\bar{U}_{m/n} = \sum_{j=n-m+1}^{n} \binom{n}{j} \bar{U}^j (1 - \bar{U})^{n-j} \quad , \tag{4}$$

where $\binom{n}{j}$ is the binomial coefficient and $\bar{U}$ is given by Eq. (1). The dominating terms are $\bar{U}_{1/2} \approx \bar{U}^2$, $\bar{U}_{1/3} \approx \bar{U}^3$, $\bar{U}_{2/3} \approx 3\bar{U}^2$, $\bar{U}_{1/4} \approx \bar{U}^4$, $\bar{U}_{2/4} \approx 4\bar{U}^3$ and $\bar{U}_{3/4} \approx 6\ \bar{U}^2$. With these relations and Eq. (1) it is easy to select a preliminary design for redundancy and test interval.

The models described and the ICARUS code have been used to calculate the average unavailability of an AFWS consisting of three redundant lines requiring the operation of either a steam turbine driven pump or one of the electrically driven pumps. The system is depicted in Fig. 1. Data obtained from the literature [8, 9, 2] for testing and repair practice as well as failure data are summarized in Table I. Assuming that the failure data for steam and electric pumps are the same, the trains are identical, each consisting of one pump and two valves in series. Data for such a train is also given in Table I, obtained from the component data [1].

Table I. Data for pumps and valves

Symbol	Pump (P) (Fail to start)	Valve (V) (Plug)	Train (P + 2V)
υ	1.4h	0.9h	2.3h[a]
q_0	1.0	1.0	1.0[b]
p	4×10^{-4}	3×10^{-4}	10^{-3}
γ	8×10^{-4}	6×10^{-4}	2×10^{-3}
ρ	4×10^{-4}	3×10^{-4}	10^{-3}
λ	1.4×10^{-6}/h	1.4×10^{-7}/h	1.7×10^{-6}/h
ω	0.02	0.02	0.02
τ	19h	7h	13h
λ_μ	4×10^{-7}/h[c]	1.4×10^{-7}/h[d]	5.4×10^{-7}/h

[a] Two valves tested simultaneously.

[b] Test override capability for some transients yields $q_0 < 1$.

[c] Due to a non-automatic start of the pump in a test.

[d] Due to a check valve not tested monthly.

The value of $\bar{U}$ of Eq. (1) with $\eta = 720$h is 6.9×10^{-3}, of which 44% is due to $\rho + \gamma$, 47% due to testing ($q_0\upsilon$) and 9% is due to random failures (λ).

Results for the system are given in Fig. 2. The three cases shown reflect the different configurations that the AFWS can take. Case I is a 1/3-system assuming that both off-site power and plant steam are available. Case II is a 1/2-system assuming the availability of electric power only, and Case III is a 1/2 system assuming neither plant steam nor off-site power. The large failure rate 4×10^{-5}/h of the diesel generators increase the unavailability of Case III compared to Case II. The results in Fig. 2 indicate that diesel generators could be tested more frequently than pumps, and that a third diesel generator would significantly improve the availability. Staggered testing seems to be somewhat better than the other testing schemes. Besides, staggered testing provides recovery time for maintenance personnel so that repeated errors are less likely to occur. Furthermore, different testing or supervisory personnel can be used in different tests more easily with staggered than with nearly simultaneous testing schemes.

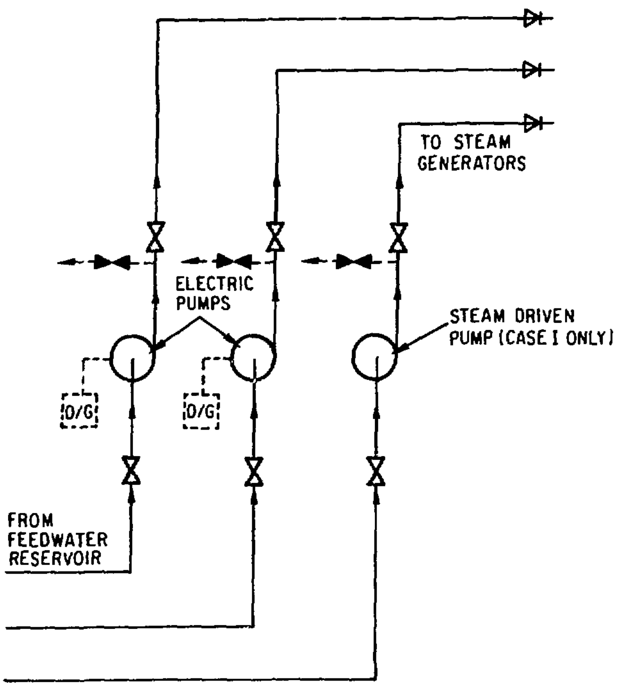

Fig. 1 - Simplified Flow Diagram of an Auxiliary Feedwater System

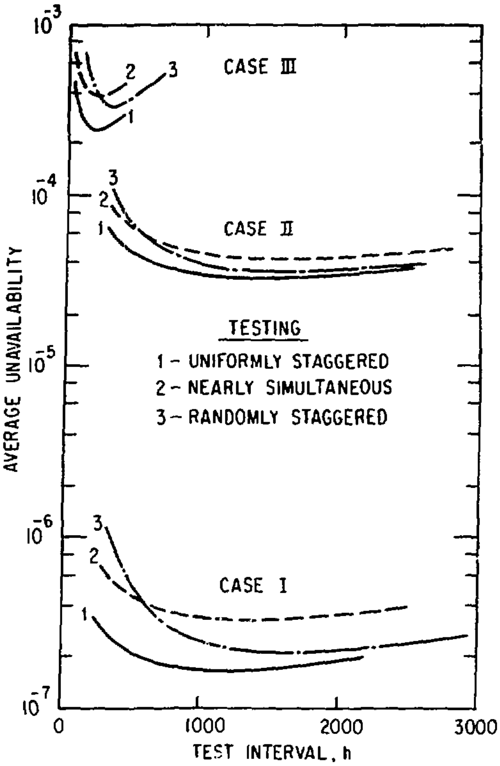

Fig. 2 - Average Unavailability vs. Test Interval in Cases I, II and III with Three Testing Schemes

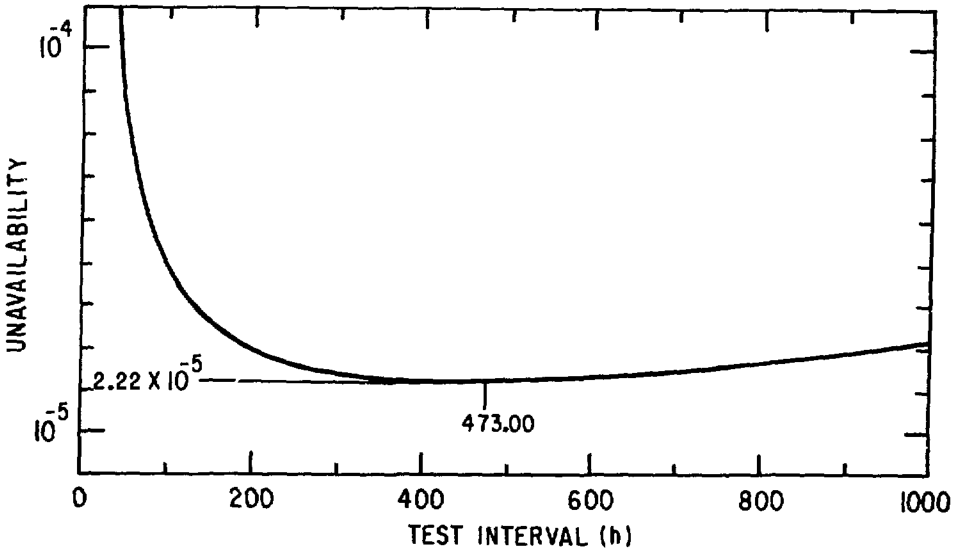

Fig. 3 - Average Unavailability of an AFWS as a Function of Test Interval

When the probabilities of different cases are P_I, P_{II} and P_{III}, the resultant unavailability is the weighted average of the unavailabilities of the three cases. This is given in Fig. 3 for the staggered testing scheme when $P_I = 0.7$, $P_{II} = 0.27$ and $P_{III} = 0.03$. About 59% of the minimum unavailability 2.2×10^{-5} at $\eta = 473h$ is due to Case III. This again indicates the significance of the diesel generators.

UNDETECTED FAILURES

Safety systems may contain components that can not be thoroughly tested while the plant is operating. For example, a manual starting of a pump in a test does not reveal failures in the automatic starting circuit. When the rate of systematically undetected failures is λ_μ, the instantaneous unavailability of a component is

$$V(t) = 1 - e^{-\lambda_\mu t} + U(s)e^{-\lambda_\mu t} \approx \lambda_\mu t + U(s) \ , \ 0 \leq t \leq T \ , \tag{5}$$

where U(s) is the periodic unavailability defined in the previous section and s is the relative time counted from the latest test prior to t. The total time T is one year if a thorough inspection is carried out annually. Assuming that $\lambda_\mu T < 0.1$, the average unavailability is

$$\bar{V} = \bar{U} + \lambda_\mu T/2, \tag{6}$$

where $\bar{U}$ is given by Eq. (1). For redundant systems with n parallel trains it can be shown that, with the random testing scheme, the average unavailabilites are [11]

$$\bar{V}_{1/n} = \bar{U}^n[1 + R_{1/n}(x)], \quad 1 \leq n \quad , \tag{7}$$

where

$$R_{1/1}(x) = x \ , \tag{8}$$

$$R_{1/2}(x) = 2x + \frac{4}{3} x^2 \ , \tag{9}$$

$$R_{1/3}(x) = 3x + 4x^2 + 2x^3 \ , \tag{10}$$

$$R_{1/4}(x) = 4x + 8x^2 + 8x^3 + \frac{16}{5} x^4 \ , \tag{11}$$

and

$$x = \lambda_\mu T/(2\bar{U}) \ . \tag{12}$$

For the other redundancies

$$\bar{V}_{2/3} = 3 \ \bar{V}_{1/2}, \ \bar{V}_{2/4} = 4 \ \bar{V}_{1/3}, \ \bar{V}_{3/4} = 6 \ \bar{V}_{1/2} \ . \tag{13,14,15}$$

The terms $R_{m/n}(x)$ are the undetected failure contributions to the system unavailability. These polynomials are presented in Fig. 4 as functions of the ratio x. This figure indicates that the higher the redundancy (n), the more sensitive the system is to undetected failures. In the region $x < 0.05$ the undetected failure contribution can be neglected.

COMMON-MODE FAILURES

A common-mode failure is defined as an event which causes a coincidence of failed states in two or more separate trains of a redundancy system. Let $\Gamma_r(t)$ be the unavailability of exactly r trains due to a common-cause failure. The average unavailability of an m/n-system is [11]

$$\bar{W}_{m/n} \simeq \bar{V}_{m/n} + \sum_{r=2}^{n} \bar{\Gamma}_r \bar{V}_{m/n-r} , \tag{16}$$

where $\bar{\Gamma}_r$ is the average of $\Gamma_r(t)$, $\bar{V}_{m/n-r}$ was defined in the previous section for r = 2,..., n-m, and $\bar{V}_{m/n-r} \equiv 1$ for $r > n-m$.

Several compilations of common-mode failure data indicate that as much as 20 to 30 per cent of all failures are caused by common-mode failures [9, 10]. Relatively little attention has been paid on how long these failures remain undetected in the system, although this information is very essential for the availability of a safety system. By analyzing the common-mode events compiled in Ref. [9] for AFWS's, it is possible to estimate both the frequencies and the residence times of the common-mode failures. The experience collected in years 1974 to 1976 indicates that $\bar{\Gamma}_2$ could be as large as 10^{-2} and $\bar{\Gamma}_3$ larger than 10^{-3}. The main reason for smaller $\bar{\Gamma}_3$ is diversity, i.e. the third AFWS pump is driven by a steam turbine rather than electric power. For a 1/3-system Eq. (16) yields

$$\bar{W}_{1/3} = \bar{V}_{1/3} + \bar{\Gamma}_2\bar{V} + \bar{\Gamma}_3, \tag{17}$$

where $\bar{V}_{1/3}$ and $\bar{V}$ are given by Eqs. (7) and (6), respectively.

The effects of both undetected failures (λ_μ) and common-mode failures are illustrated in Fig. 5 for the 1/3-system of Case I discussed earlier, assuming different values of $\bar{\Gamma}_3$ while $\bar{\Gamma}_2 = 0$. The numerical values of $\bar{\Gamma}_3$ in Fig. 5 are valid for the test interval of 720h. Two curves are presented for each value of $\bar{\Gamma}_3$, a dotted one valid when $\bar{\Gamma}_3$ is a constant independent of the length of the test interval, and a solid one valid when $\bar{\Gamma}_3$ is proportional to the test interval. The latter is the case if common mode failures occur at random times (not associated with testing or true demands) and are detectable i.e. revealed by periodic tests. It appears that both undetected failures and common-mode failures decrease the significance of optimization by increasing redundancy. For example, the value of $\bar{\Gamma}_3 = 3 \times 10^{-5}$ in a 1/3-system brings the unavailability up to the level of a 1/2-system, as can be seen by comparing Figs. 2 and 5. Optimization by controlling the test interval is possible for a certain class of common-mode failures.

CONCLUSIONS

Analytical models and computer codes have been developed to calculate the unavailabilities of redundant safety systems with components tested periodically. Common-mode failures and undetected failures as well as a variety of testing, repair and hardware failures have been taken into account such that the parameters can be estimated from existing data. The contributions of various failure modes and testing schemes have been illustrated by examples.

Applications to an AFWS of a typical PWR of a 3-loop design indicate that diesel generators with high failure rates probably are the most critical com-

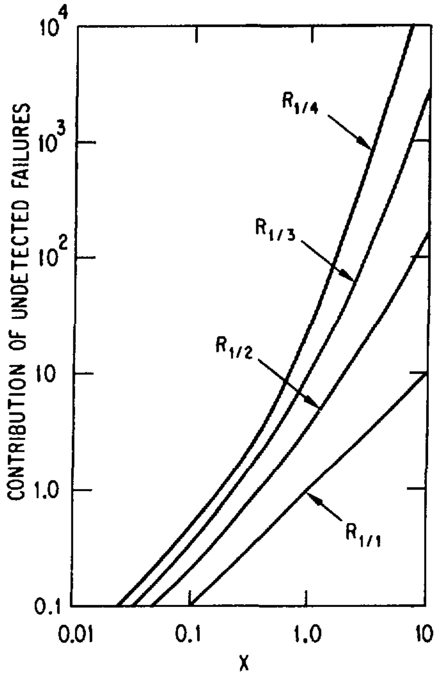

Fig. 4 - Contributions $R_{m/n}$ (x) of Undetected Failures vs. Ratio $x = \lambda_{\mu}T/(2\bar{U})$.

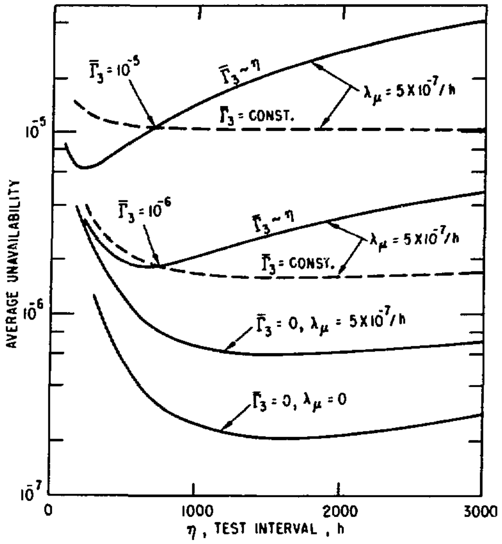

Fig. 5 - The Effects of Undetected Failures and Common-Mode Failures on the Average Unavailability of a 1/3-system (Case I, Random Testing Scheme).

ponents. Using three diesel generators rather than two would considerably improve the availability, given that the common-mode contribution is small. However, current data indicate that common-mode failures caused by human errors in design, operation and maintenance most likely are the main obstacle for improving the system availability by optimizing the redundancy and testing. Similar conclusions can probably be drawn for other safety systems consisting of pumps and valves and associated control circuits. Major efforts should be directed to eliminating common-mode failures caused by human errors in engineering procedures.

ACKNOWLEDGMENTS

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FESSENHEIM PLANT

PROBABILISTIC STUDY OF THE SPURIOUS DILUTION OF THE CHEMICAL AND VOLUME CONTROL SYSTEM (C.V.C.S.)

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ABSTRACT

The system analyzed was the Chemical and Volume Control System (C.V.C.S.) which cannot be studied separately from the Water Make-up System.

The operational modes which have been studied are :
- continuous power operation ;
- cold shutdown for the maintenance of the steam generators.

A Failure Modes and Effects Analysis (F.M.E.A.) was carried out in conjunction with a Fault Tree Analysis. The quantitative analysis was performed using a computer code.

For both operational modes, the occurrence probability of dilution has been evaluated for each dilution level and compared with the results of a survey of spurious dilution incidents in U.S. P.W.R. nuclear power plants.

The spurious dilution risk is relatively high but the risk of uncontrolled dilution is rather low indeed considering the alarms, the protection system channels and the time available for the operator to stop the dilution.

INTRODUCTION

Electricite de France (E.D.F., French Utilities) has been performing systems probabilistic reliability studies for its nuclear power plants, since 1975.

This paper contains the major findings of the probabilistic study on spurious dilution of the chemical and volume control system (C.V.C.S.) for the Fessenheim plant [1]. This study was performed in the framework of the Project Work Statement (P.W.S.) 2-12 of the Research and Development Agreement between C.E.A. (French Atomic Energy Commission), E.D.F., FRAMATOME and WESTINGHOUSE (WEREAS)

The consequences of an uncontrolled spurious dilution depend on the plant operational mode. The resulting risks are the following :

- during shutdown, uncontrolled return to core criticality ;
- during continuous power operation, boiling crisis which could be caused by high nuclear flux.

The system analysed was the Chemical and Volume Control System which cannot be studied separately from the Water Make-up System.

C.V.C.S. PRESENTATION

As regards P.W.R. plants, the French R.C.V. (Reacteur Contrôle Chimique et Volumétrique) and R.E.A. (Reacteur Appoint Eau et Bore) systems taken together are the counterpart of both the Chemical and Volume Control System and the Water Make-up System. Considered separately, the R.C.V. System is equivalent to the C.V.C.S. except for the recycle process and the boration line. The R.E.A. System corresponds to the Water Make-up System plus the C.V.C.S. boration line.

The simplified diagrams of the R.C.V. and R.E.A. Systems are presented in figures 1 and 2.

Functions

The R.C.V. and R.E.A. Systems must perform the following functions :

- Functions common to the R.C.V. and R.E.A. Systems :
  - . control the primary system reactivity
  - . maintain the required water inventory in the reactor coolant system
  - . modify the nature and concentration of the gases dissolved in the primary water
  - . add chemicals.
- Specific functions of the R.C.V. System :
  - . maintain the R.C.S. (Reactor Coolant System) water inventory within the allowable pressure range during reactor shutdown and start-up
  - . maintain adequate primary water quality
  - . provide auxiliary pressurizer spray
  - . provide seal and cooling water to the reactor coolant pump n°1 seals
  - . provide high pressure injection flow to the reactor coolant system (as a matter of fact, this function is included as part of those of the Safety Injection System).
- Specific functions of the R.E.A. System :
  - . prepare the boric acid concentrated solutions
  - . provide reactor coolant drain tank spray
  - . provide seal water to the reactor coolant pump seals.

Operation

R.C.V. System

Reactor coolant enters the Chemical and Volume Control System. After temperature and pressure reductions, the primary fluid is purified before entering the volume control tank. The charging pumps take suction from the volume control tank, return the coolant to the Reactor Coolant System and provide seal water injection to the reactor coolant pumps.

R.E.A. System

The R.E.A. System essentially includes two units. The first one is used to batch, store and deliver boric acid solutions. The second one is used to store and deliver primary water. Selection of either the boration, dilution or the automatic make-up mode is performed by the reactor make-up control.

SCOPE OF THE STUDY

After examination of all unit operational modes we decided to only study :
- continuous power operation because it is the most common operational mode and it is very similar to hot shutdown from the spurious dilution point of view
- cold shutdown for the maintenance of the steam generators because during this mode the primary coolant inventory is minimum.

Moreover our attention was focused on the possible initiating events attributable to the R.C.V. and R.E.A. systems and which might lead to a spurious dilution of the reactor coolant.

The accident sequences initiated by a potential dilution that might lead either to the boiling crisis, in the case of continuous power operation, or to core criticality in the case of reactor cold shutdown, were not thoroughly studied.

METHODOLOGY

We carried out a Failure Modes and Effects Analysis (F.M.E.A.) for the R.C.V. and R.E.A. Systems in conjunction with a Fault Tree Analysis. For these analyses, we examined many documents (Final Safety Analysis Report, operating procedures) and performed in situ inspection of the components.

A Failure Mode and Effects Analysis must be, from our point of view, previous to any system reliability analysis. It represents a very useful aid for the analyst to try to realize a complete work.

The Fault Tree Analysis was very useful and well-fit to this type of problem. But our experience during these last years has shown to us that this method was sometimes uneasy to use.

After this qualitative analysis, the quantitative analysis was performed using computer codes. The first code "CASSIS" [2] enables the transformation of fault trees into reliability block diagrams. The second code "FIAB C" [3] which can be coupled with the first one uses as input reliability block diagrams and data concerning the components and gives the estimated value of the studied risk with the relevant confidence interval as well as the importance factors for each component [4].

The data used in the calculations were derived from the operational experience of Electricite de France in nuclear power plants [5] and from the specialized literature.

MAJOR FINDINGS

For both operational modes, the occurrence probability of a dilution has been evaluated for each dilution level.

We have drawn a distinction between two cases depending on whether the volume of the water entering the reactor coolant system is limited or not. For each critical component when considering the risk of spurious dilution, we have specified the maximum dilution level, the time available for the operator to start a corrective action and the importance of the component from the final results point of view.

Table I gives the probability of dilution during continuous power operation.

Table II gives this probability during the reactor cold shutdown for maintenance operations on the steam generators (5 days a year).

COMPARISON OF THE RESULTS WITH THE U.S. P.W.R. OPERATIONAL EXPERIENCE

Main differences between the C.V.C.S. in Fessenheim and the same system in U.S. Westinghouse plants

From the spurious dilution point of view, the R.C.V. and R.E.A. systems of the Fessenheim plant and the C.V.C.S. of the U.S. plants are very similar. The main differences are :

- the concentration of the boric acid solution (most of U.S. plants use boric acid to a concentration of 12% by weight, the Fessenheim plant as opposed uses 4% by weight)
- the operational mode of water make-up pumps and boric acid pumps
- the existence or not of continuous monitoring of the boron concentration in the reactor coolant system.

For this comparison we used Final Safety Analysis Reports (F.S.A.I ` prepared for U.S. plants.

Survey of spurious dilution incidents in U.S. power plants (up to august 1978)

All these incidents occurred during cold shutdown. Indeed, a spurious dilution, when the containment integrity is maintained, is not regarded as an abnormal occurrence. These incidents were listed in the Nuclear Power Experience (N.P.E.) and Nuclear Safety Analysis Center (N.S.I.C.) files.

Six incidents of this type were identified. Three of them were due to human error. Among those, two were connected with the demineralizers.

Two others were induced by a failure of the reactor make-up control. The last one was caused by a leakage through a steam generator.

The survey of these incidents led to a probability estimate for spurious dilution of approximately 0.4 in one year of cold shutdown.

CONCLUSIONS

One common characteristic of all dilution risks for almost every operational mode, is the essential part played by human error.

The spurious dilution risk looks relatively high. So we have examined the different alarms and automatic actions which are initiated in case of a dilution for the different operational modes, and the time available for the operator to stop the dilution. Table III gives the time interval evaluations for the operator to stop the dilution after the initiation of the first alarm and before returning to criticality.

In conclusion, although the occurrence probability of a spurious dilution is relatively high, we can reasonably think that the risk of uncontrolled dilution is rather low indeed, in all cases considered, if the alarms and the protection system channels (Reactor Protection System) operate properly, the operator has about an hour to find the cause of the dilution or to initiate a boration.

TABLE I
RESULTS OBTAINED FOR THE CONTINUOUS POWER OPERATION

Dilution level	All flow rates (water volume limited or not)	All flow rates (unlimited water volume)	Flow rates $\geq 10\ m^3/h$
Estimated value	0.7/year	0.4/year	0.3/year 0.2/year**
Error factor	1.4	1.9	2.1 2.3**
Main components in the dilution risk/Time available for the operator to stop the dilution*/ Dilution flow rate/ Probabilistic importance of the component.	1 - R.C.V. demineralizers and associated valves (1) /10h/5 m^3/h/0.35 2 - Three-way valve R.C.V 26 VP (1)/∞/13.6 m^3/h/0.17 3 - Mode selector switch of R.E.A. (1)/2h/ 27 m^3/h/0.17 4 - Preset quantity of primary water make-up (1) 2 h/27 m^3/h/0.17	1 - Mode selector switch of R.E.A. (1) 2h/27 m^3/h/0.28 2 - R.C.V. demineralizers and associated valves (1)/ 10 h/5 m^3/h/0.24 3 - Preset quantity of primary water make-up (1)/ 2 h/27 m^3/h/0.21 4 - Flowmeter R.E.A. 14 MD/2h/27 m^3/h/0.07	1 - Mode selector switch of R.E.A. (1)/2h/ 27 m^3/h/0.40 (0.44)** 2 - Preset quantity of primary water (1)/2h/27 m^3/h 0.34 (0.28)** 3 - Flowmeter R.E.A. 14 MD/2 h/27 m^3/h/0.11 (0.09)**

TABLE II
RESULTS OBTAINED DURING COLD SHUTDOWN FOR MAINTENANCE ON THE STEAM GENERATORS

Dilution level	All flow rates (water volume limited or not)	All flow rates (unlimited water volume)	Flow rates$\geq 10\ m^3/h$
Estimated value	1.5×10^{-2}/year	8.3×10^{-3}/year	3.1×10^{-4}/year
Error factor	8	7	3.5
Main components in the dilucion risk/Time available for the operator to stop the dilution/Dilution flow rate/Probabilistic importance of the component	1 - R.C.V. demineralizers and associated valves (1)/ 7 h/5 m^3/h/0.73 2 - R.E.A. line for chemical additions (1)/ 34 h/1 m^3/h/0.20	1 - R.C.V. demineralizers and associated valves (1)/7 h/5 m^3/h/0.36 2 - R.E.A. line for chemical additions (1)/34 h/1 m^3/h/0.36	1 - Seal water heat exchanger R.C.V. 03 RF /1 h 40/20 m^3/h/0.48 2 - Non-regenerative heat exchanger 02 RF/2 h 10/ 2 h 10/15 m^3/h/0.48

(1) human error.

* During hot shutdown, time must be divided by two.

** The assumption was made that for some components a failure will occur in 50% of cases which will lead to a reduction of the water volume sent to the primary coolant below the required level.

TABLE III

EVALUATION OF TIME INTERVAL BETWEEN THE APPEARANCE OF THE FIRST ALARM AND THE RETURN TO CRITICALITY AND AVAILABLE ALARMS

Continuous power operation (automatic control)	Hot shutdown	Cold shutdown for maintenance on steam generators
1 h 40 mn	1 h 10 mn	1 h
- Rod cluster control assemblies insertions - Low position of D rod cluster control assembly - Very low position of D rod cluster control assembly - Scram	- low boron concentration - High neutronic flux during shutdown - Scram	- low boron concentration - High neutronic flux during shutdown - High level in the reactor coolant system

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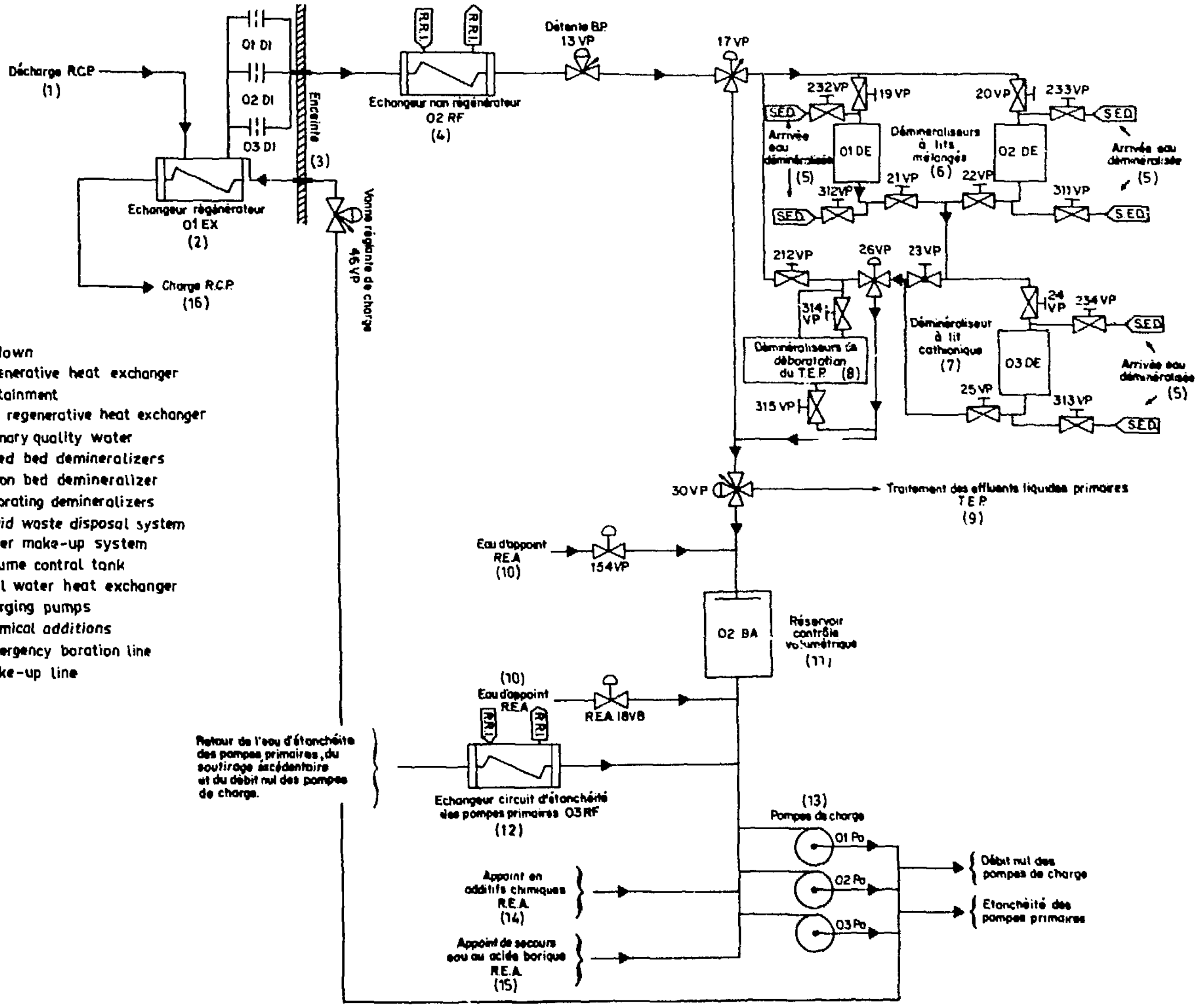

(1) Letdown
(2) Regenerative heat exchanger
(3) Containment
(4) Non regenerative heat exchanger
(5) Primary quality water
(6) Mixed bed demineralizers
(7) Cation bed demineralizer
(8) Deborating demineralizers
(9) Liquid waste disposal system
(10) Water make-up system
(11) Volume control tank
(12) Seal water heat exchanger
(13) Charging pumps
(14) Chemical additions
(15) Emergency boration line
(16) Make-up line

FIGURE 1_SIMPLIFIED DIAGRAM OF THE R.C.V. SYSTEM

(1) Boric acid intake to a concentration of 4% by weight
(2) Boric acid transfer pumps
(3) Flowmeter
(4) Boric acid blender
(5) Normal make-up
(6) Emergency make-up
(7) Chemical additions
(8) To the charging pumps
(9) Primary quality water pumps
(10) Containment
(11) Chemical mixing tank
(12) To the reactor coolant system

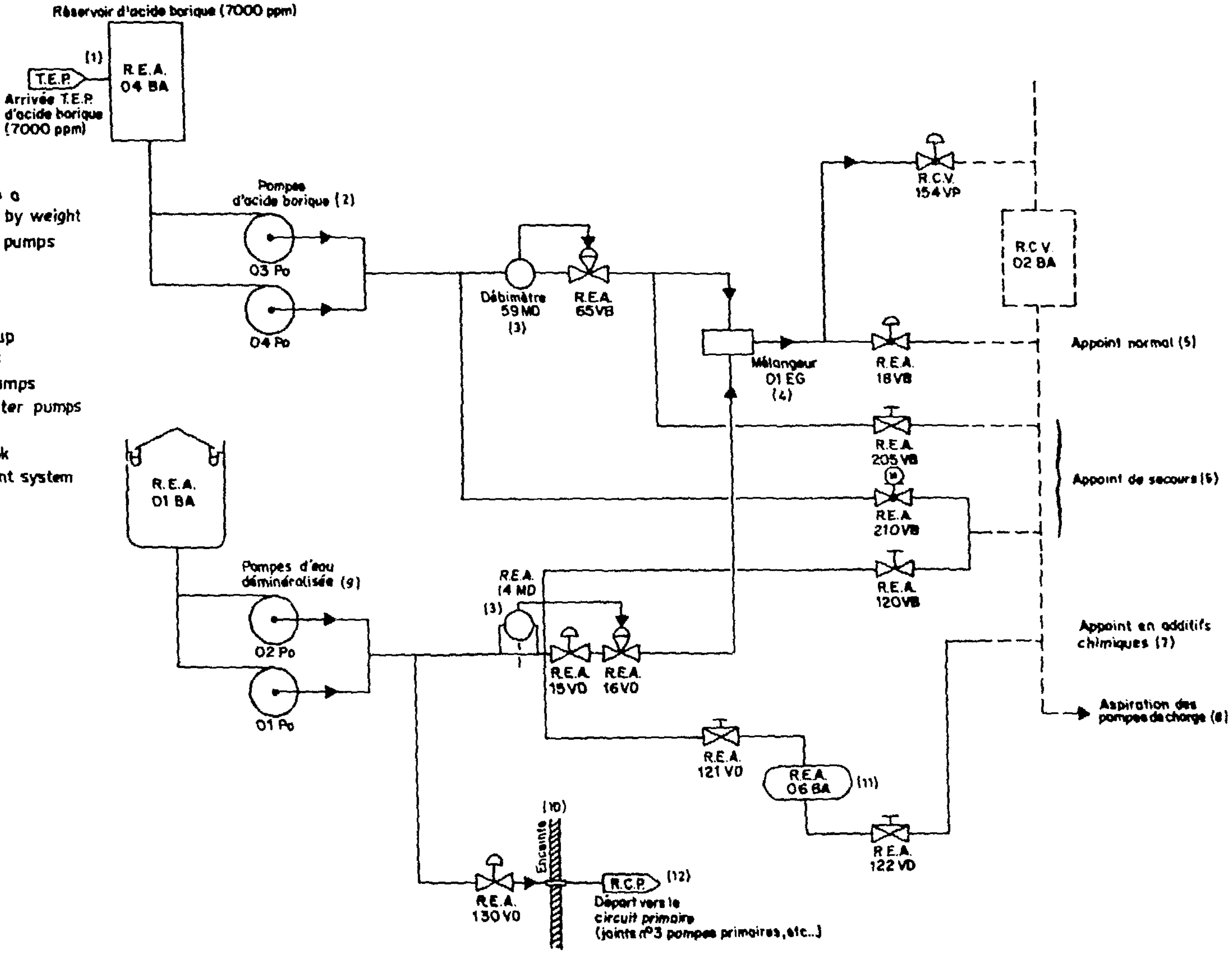

FIGURE 2 _ SIMPLIFIED DIAGRAM OF THE R.E.A. SYSTEM

RELIABILITY STUDY OF THE INTEGRATED CONTROL SYSTEM

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ABSTRACT

A study was performed to determine the reliability of the Integrated Control System (ICS) provided with the Babcock and Wilcox NSS. The study addressed both the field performance of the ICS and its potential failure modes. The study reveals that the ICS is a reliable system that has performed well and that failure of the system will not provide protection system challenges that exceed the design capability of the plant.

INTRODUCTION

The Integrated Control System (ICS) is provided with The Babcock & Wilcox NSS and is responsible for proper co-ordination of the reactor, steam generator feedwater, the main turbine, and the turbine bypass valves under all operating conditions. The ICS is unique among NSS control systems in its ability to provide fully automatic plant control both during normal plant maneuvers as well as during upsets resulting from many abnormal conditions. The purpose of this paper is to describe the results of a reliability analysis performed on the ICS. The analysis was documented by the report "Integrated Control System Reliability Analysis," BAW-1564, August 1979. The report consisted of two distinct parts:

1. An operating experience survey that assessed the performance and reliability of the ICS after 36 reactor years of operation, and

2. A failure modes and effects analysis that identified potential areas for ICS improvement.

ICS OPERATING EXPERIENCE

Transient event data were obtained by B&W engineers from each operating plant. The data base included information from Oconee Units 1,2, and 3, TMI-1, and 2, Crystal River, ANO-1, Rancho Seco, and Davis Besse. The event data obtained from the operating plants included reactor trip writeups, the control room logs, licensee event reports (LERs), transient records (where available), allowable operating transient cycle (ATOC) data (where available), and records of maintenance and repair from the reactor station instrument shop records.

EVALUATION OF DATA

Evaluation of the data available indicated that the bulk of the information dealt with reactor trips and the sequence of events leading to the reactor trip. Except for startup testing little information was readily available about successful "no-trip" sequence of events.

After an initial review of the transient data, and in view of the type of data available, it was clear that the data could be sorted and analyzed on the basis of initiating events. On this basis the operating plant transient data was sorted into six major categories for the purposes of analysis and evaluation:

A_1: ICS Response: Events in which ICS response permitted or led to a reactor trip. The operating plant data indicated several events in which the ICS (a) participated in feedwater oscillations, (b) did not respond properly to an upset, or (c) initiated an upset that led to a reactor trip.

A_2: ICS Internal Failure: Malfunctions or failures within the ICS cabinets, excluding external power supplies, that led to a reactor trip.

B: Input Failures: Malfunctions or failures of signal inputs (including power supplies) to the ICS that led to reactor trips.

C: Actuator Failures: Malfunctions or failures of devices controlled or actuated by the ICS which led to reactor trips.

D: Operator/Technician Action: Reactor trips caused by operator, technician, or maintenance action. This category included planned reactor trips.

E: Other Plant Events (Usally BOP): Failures or malfunctions in other Power System Equipment that led to reactor trips. This category included steam and feedwater equipment, station power supplies, and spurious RPS trips.

Figure 1 is a graphic presentation of the operating experience survey. This figure divides the 310 reactor trips that had been experienced on B&W plants into the six categories of initiating events discussed above.

CONCLUSIONS BASED ON OPERATING DATA

Based upon the study of field performance the following conclusions were reached:

1. No events were identified in which the ICS caused the RPS design bases to be violated.
2. B&W concluded that the ICS design concept, i.e., coordinated control of reactor power, feedwater, and turbine power, runback features on upset events, and crosslimits, is a correct and proper control strategy for availability and safety.

3. The ICS hardware performance contributed to only 1.9% of the reactor trips.

4. The NNI/ICS power sources (external to ICS cabinets) had been vulnerable to single failures and human errors that led to reactor trips and sometimes plant overcooling.

5. ICS/power system tuning is an important feature of plant availability. While the level of system response to upset transients and system interactions varies from plant to plant, B&W believes that tuning can be improved at each station.

6. Failures and malfunction of the actuated equipment, including control rod and power supply systems, FW valves, feed pump drives, and the turbine controls contribute greatly to plant unavailability. Periodic testing and planned improvements in equipment can contribute to availability and enhance operations during upset conditions.

7. Inputs to the ICS that show a relatively significant failure rate should be improved to enhance plant availability.

8. The BOP power systems equipment has contributed to many reactor trips and plant unavailability. These systems should be reviewed for availability and reliability improvements, particularly the main FW and condensate system.

FAILURE AND EFFECTS ANALYSIS

The failure modes and effects analysis (FMEA) was performed to identify sources of transients initiated by the ICS and to define potential areas for improvement that could reduce the frequency of these transients. The FMEA also determined whether an ICS failure could cause a failure mode whereby the safety systems would not protect the reactor core. The emphasis was on analyzing ICS failures that could affect the feedwater system, emergency feedwater system, or the pressurizer thus creating a challenge to the PORV, safety valves, or RPS/ESFAS.

METHOD OF PERFORMING THE FMEA

To perform the ICS FMEA, each input and output to a typical ICS was assumed to fail "high" or "low", one at a time, to determine its effect upon the NSS. For inputs, high was selected as the maximum output of the transmitter and low as the minimum. For ICS outputs, high was chosen as the output signal that fully opened valves, caused pumps to attain maximum speed, pulled control rods, etc. while a low output causes the inverse of these actions; valves close, pumps go to the low speed stop, control rods insert, etc. Since each operating B&W NSS and ICS is slightly different, the Rancho Seco plant was chosen as a representative design. The FMEA is therefore specific to that plant. However, because of the close functional similarity of plants the results of the study are equally applicable to all 177-FA B&W plants. Of course, when analyzing the effects of an input or output failure, results are dependent on how the input or output fails, the NSS condition when failure occurs (time in core life, power level, etc.), and a host of other variables. Since it was impossible to analyze all cases in a reasonable time frame, an attempt was made to study the worst case: step failures of inputs and outputs high or low at that NSS power level where failure causes the most drastic transient.

The FMEA also considered the results of failing various functional blocks internal to the ICS. For this purpose a functional drawing of the ICS was prepared and is shown in Figure 2. Using this functional description internal failures, designated as high or low in a manner similar to output failures, could be postulated and their effect upon the NSS determined.

METHODS OF DETERMINING FAILURE EFFECTS ON THE NSS

The ideal method of determining failure effects upon the NSS is of course to fail the variable or component on a real plant. This is obviously an unrealistic and undesirable method. There were, however, a few failures, notably failure of the RC flow inputs to the ICS and ICS power supply inputs, that have occurred on operating plants. In those cases field results were factored into the tables. In most instances no operating plant results were available since field failures are not common. For those failures a combination of hybird computer simulation results plus in-depth understanding of the ICS and NSS was used to determine the worst credible NSS effect. The hybird computer simulation used in the study is the latest version of the POWER TRAIN IV computer code, which simulates a B&W 177-FA NSS and was made specific to the Rancho Seco plant. The simulation includes the pressurized water reactor, pressurizer, primary loop piping, parallel OTSGs, FW system, turbine bypass system, turbine generator, RPS, and ICS.

DISPLAYING FMEA RESULTS

Table I is an example of the tables prepared to show the results of failing the inputs, outputs, and internal modules of the ICS. A brief discussion of the column headings follows:

Sheet and Item No.	This references the input or output failed to the complete listing given in the Rancho Seco ICS I/O list, or to the functional drawing.
Input	A verbal description of the input, output, or functional block under consideration.
Failure Mode	Describes how the variable was assumed to fail.
Effect on NSS	This table entry describes briefly the transient expected if the variable fails. An attempt was made to describe the least desirable result.

Reactor Trip	States whether or not a reactor trip is expected when the failure occurs.
Remarks	Gives general comments on the failure.

CONCLUSIONS DRAWN FROM FMEA

The overall conclusion of the FMEA is that the reactor core remains protected throughout any of the ICS failures studied. It was also found that many postulated single failures could cause a NSS transient that might result in reactor trip some time in core life and at some power level. The bulk of transients initiated by control system failures can be overcome by timely operator intervention and even without operator action most transients are terminated by reactor trip. The FMEA did uncover three general categories of failures that required either operator action or the initiation of additional systems such as high pressure injection or auxiliary feedwater to terminate the ensuing NSS transients. A brief discussion of each category follows:

1. The FMEA indicated that an inadvertently opened or stuck open turbine bypass valve could result in overcooling. (The plant data do not show a significant frequency of turbine bypass malfunctions, however.)
2. The FMEA also indicated that an inadvertently opened or stuck open main feedwater startup valve could result in steam generator overfill and overcooling.
3. The FMEA identified feedwater pump speed control failure to both feedwater pumps as the only postulated failure that could adversely affect feedwater control to both steam generators after a reactor trip.

CONCLUSIONS DRAWN FROM THE RELIABILITY STUDY

The FMEA clearly pointed out that the ICS, either through its own failure or through its response to failed instrumentation or abnormal plant conditions, can initiate transients . There is no evidence, however, that the ICS causes frequent or severe challenges to the plant protection systems (PPS) or that these challenges exceed the PPS capability. On the contrary, examination of field performance data reveals that only a small number of ICS failures have caused reactor trips. The field data, supported by conversations with plant operators, demonstrate that the ICS is both reliable and failure tolerant to a significant degree.

TABLE I EXAMPLE OF A FMEA DISPLAY TABLE

Sheet & Item No.	Input	Failure Mode	Effects on NSS	Reactor Trip	Remarks
1-19	Turbine Header Pressure	1200 psia	Condenser dump & atmospheric dump valves go full open. Turbine throttle valve opens slightly and turbine transfers to manual. Steam pressure decreases, MWE drops, and an RC pressure trip normally results.	Probable	Steam pressure control after a reactor trip is available by H/A condenser dump valve control.
		600 psia	Turbine throttle valve closes to maintain setpoint steam pressure and this causes actual steam pressure to increase, which causes trip of the reactor due to high RC pressure. Satisfactory secondary steam pressure control after reactor trip via the steam line safety valves.	Probable	Not problem after a reactor trip.
1-20	S.G. Outlet Pressure, Loop A	1200 psia	S.G.-A BTU limits cause partial loss of feed flow to S.G.-A. Loop A bypass valves open. MW electric tracks down. Loop A bypass remains open after reactor trip.	Probable	SGA steam pressure can be restored by using block valves to halt bypass flow.
		0 psia	No effect on NSS except that the ICS BTU limit will be higher than normal (no problem unless a subsequent overfill incident that might depend on the BTU limits).	Not expected	
1-21	S.G. Outlet Pressure, Loop B		Results are the same as for Loop A.		

Figure 1. ANALYSIS OF 310 REACTOR TRIPS

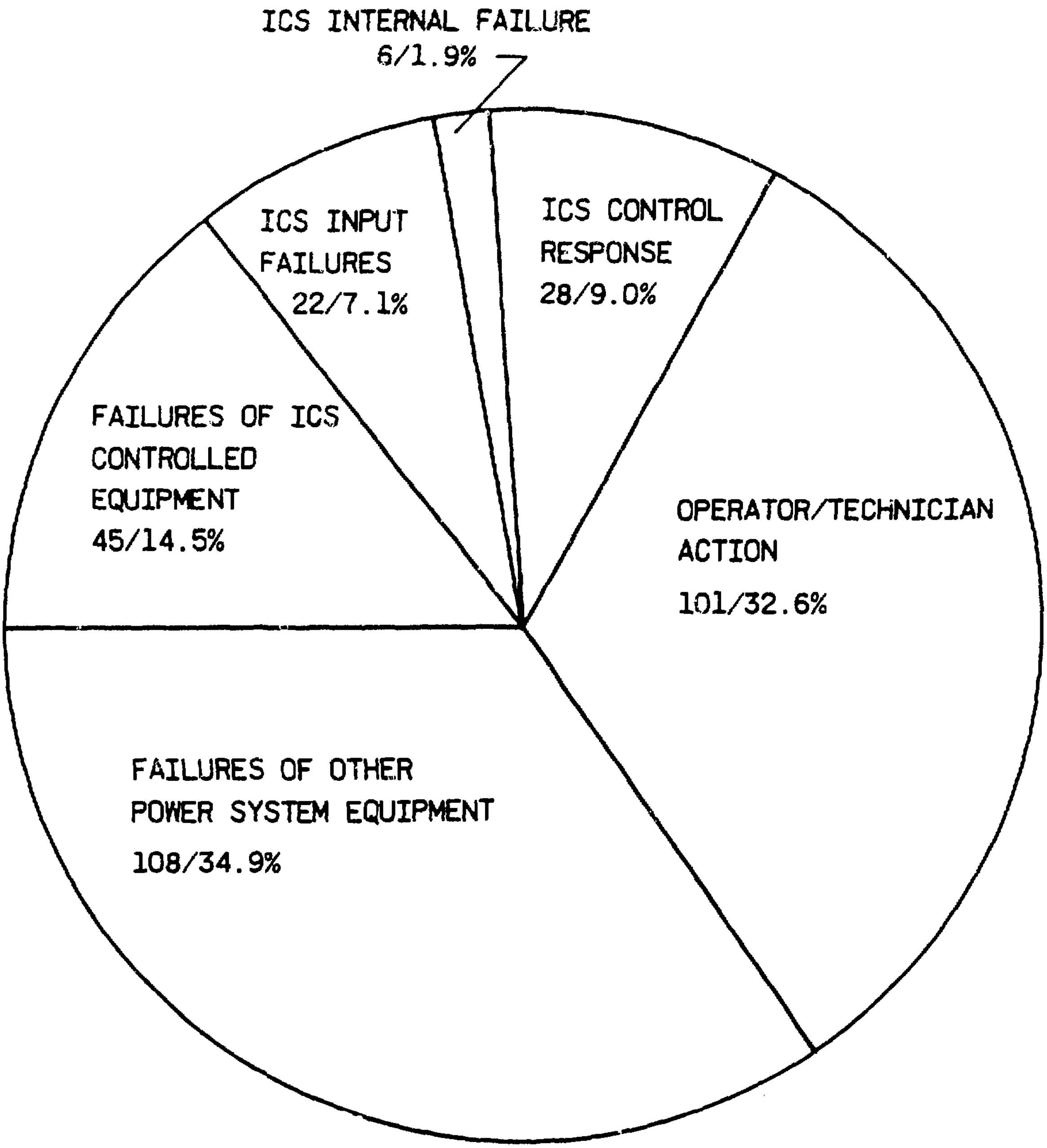
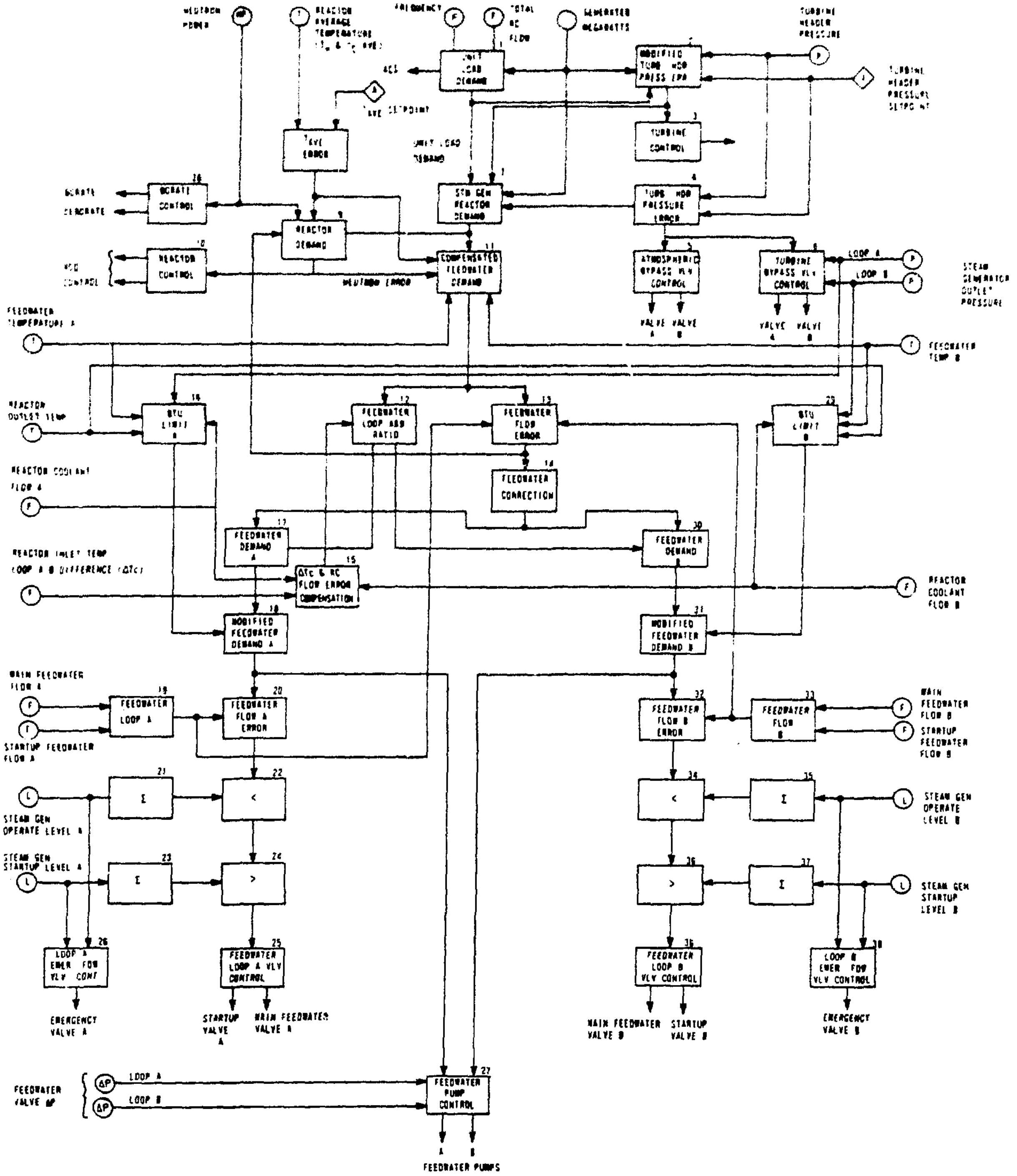


Figure 2. ICS FUNCTIONAL BLOCK DIAGRAM



VERIFICATION AND QUALIFICATION OF THE RETRAN COMPUTER CODE

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ABSTRACT

Users of a large computer code expect, (1) the code to be free of errors to the greatest extent possible and (2) the code to be able, with appropriate input, to give reasonable agreement with experimental data when applied within theoretical limitations. In the RETRAN project, a significant effort has been directed to meeting these expectations.

The first step, code verification, is best accomplished by very strict procedures during the actual coding. However, RETRAN is based on an existing RELAP code and, as a result, only complete model additions to the code were subjected to close scrutiny during the coding phase. Members of the EPRI/Utility Working Group exercised the code with a great variety of problems, and helped to identify errors in the code.

The qualification of the code is actually a measure of how well the code can analyze problems of interest. The EPRI/Working Group, EPRI and EPRI contractors have performed a number of analyses and compared results with experimental data. The code verification for RETRAN and the approach used to qualify the code are discussed in this report, along with some background information on the development of the code.

BACKGROUND OF RETRAN

The RETRAN computer program is the result of an extensive code development effort sponsored by EPRI. The effort was initiated in 1975 in response to the utility need for a more realistic appraisal of the blowdown phase of the design basis Loss of Coolant Accident (LOCA), and EPRI's need to evaluate relevant supporting experiments. The computer program, then denoted as RELAP/E, was designed to be general enough to analyze both boiling water reactors (BWRs) and pressurized water reactors (PWRs) for either large or small break LOCAs. During the first quarter of 1976, the Nuclear Safety Analysis Task Force identified the pressing need of the utilities to analyze the non-LOCA condition events for PWRs (Types I, II and III) and BWRs (Types I through VIII). In response to this request, EPRI obtained from Energy Incorporated the RETRAN system analysis submodules. Thus the RETRAN code package stems from the development of two separate code packages, RELAP/E and RETRAN. Both of these codes were based upon RELAP4/003 update 85, released by the United States Nuclear Regulatory Commission as a portion of the Water Reactor Evaluation Model. RELAP/E was developed to provide a "best estimate" thermal-hydraulic analysis of light-water reactor systems subjected to anticipated operational transients and normal startup and shutdown maneuvers. Since both codes were based on the thermal-hydraulic differential and state equations of RELAP4, and since RELAP/E was constructed for ease of model incorporation with its semimodular and dynamic structure, the operational transient models were added as options to RELAP/E and the code name RETRAN was retained.

During this time period (late 1976) the importance of the code verification effort first became apparent to EPRI. It was estimated that a minimum of 500 hours of CDC-6600 time would be required to verify this code package, in addition to an extensive amount of manpower to set up and execute the cases. It was also apparent that, when the verification phase was complete, there would be an additional delay in implementing RETRAN on the utility computers and in training their personnel in the use of this rather sophisticated computer code package. The EPRI/Utility System Analysis Working Group, consisting of 17 utilities, was established to attempt to combine these various tasks and shorten the overall time between development and application of RETRAN. Table I gives a short summary of the overall intent of the RETRAN pre-release activity.

The first phase of this activity was completed in late 1978, and RETRAN-01 was released. This version of the code has been used to analyze separate effect experiments, small scale system effects, operational transients, and PWR LOCAs.

TABLE I

A SUMMARY OF THE INTENT OF A PRE-RELEASE OF RETRAN

1. Provide participating utilities with RETRAN so they can become familiar with and competent in its use.

2. Obtain, via utility participation, a much more thorough "debugging" of RETRAN than EPRI can provide under conventional project effort.

3. Exercise RETRAN against a wide series of problems typical of utility application.

4. Qualify RETRAN against existing plant data and other analyses.

5. Reduce the confusion associated with implementing a large computer program by inexperienced users.

6. Accumulate results of RETRAN analysis from a wide-based application effort.

RELEASABILITY CRITERIA

Because RETRAN is a computer program, confidence in, and the limitations of, the program had to be established for potential code users. There are many individual factors necessary to establish confidence in a code. Defining what is a reasonable effort, coupled with the complex concept of a large computer code, make this a formidable problem. For simplicity, let us restrict our attention to one specific part of the computer code, a particular model, and address the following questions:

1. What is the "design" function of the model?
2. What are the general limitations of the model, i.e., the limiting theoretical assumptions and the range of applicability of the data base?
3. What is the specific formulation of the model, i.e., what are the closure assumptions made and the constitutive models utilized?
4. What solution technique was utilized and how dependent is this on time steps or spatial representation?
5. What is past experience with this or similar models?

At this point the concept of "consistent application" and "extended application" must be made. An application of a model is denoted as being "consistent" if the application does not violate any of the basic assumptions made in (1) through (4) above, while an "extended" application is one that does violate any of these restrictions. The fifth item above relates to previous experience associated with the model. Some models have been used extensively with satisfactory results while others are relatively new and have been tested against a limited data base. The user must have some information regarding the confidence level of each model. For a particular model, one should expect reliable "consistent applications" and hopefully some limited "extended applications". However, any "extended application" must be recognized as highly speculative and should not be expected to produce reasonable results.

From this discussion, two requirements emerge; (1) the computer code must be completely described, and (2) the confidence level of the code must be indicated. The degree of confidence in any computer program stems from two different processes; (1) the assembly verification step in which each submodel and the entire code are quality assured, and (2) the qualification of the computer code against experimental data.

These considerations led to the following releasability criteria for RETRAN:

(1) Documentation must exist to describe
  (a) the theory and assumptions made in developing the code
  (b) the code models, logic and solution schemes
  (c) how to use the code
(2) The code must be verified to assure
  (a) the coding is correct
  (b) the solution technique is stable and convergent
  (c) the code is correctly solving the equation set programmed
(3) The code must be qualified to perform the analysis required of it by
  (a) comparison with relevant test data
  (b) comparison against other calculation techniques
  (c) assuring that all results are consistent with physical assumptions made.

RETRAN DOCUMENTATION

The first criteria for the release of a code is adequate documentation. The RETRAN program is documented in a four volume computer code manual (EPRI CCM-5, Volumes I-IV) entitled "RETRAN - A Program for One-Dimensional

Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems". Volume I - Equations and Numerics, satisfies requirement 1-a and part of 1-b as it gives a detailed description of the theory and numerics used in RETRAN. (An earlier edition of this volume appeared as EPRI NP-408 and describes some models which are not included in RETRAN-01.) The remaining part of requirement 1-b, the RETRAN code logic and detailed programming description, is given in Volume II - The Programmer's Manual. Requirement 1-c is satisfied by Volume III - User's Manual, which describes the code input, output and gives a series of sample problems to assist the user. The remaining two criteria (2-3) are directly addressed in Volume IV - Applications. The next two sections of this paper describe the extent of the RETRAN verification (Criteria 2) and summarize the qualification (Criteria 3) presented in Volume IV.

RETRAN VERIFICATION

In the development of any product the question inevitably arises as to how much quality assurance is required to produce an acceptable product. This question is especially difficult for computer codes. The basic consideration is not related to the usefulness or accuracy of the product, but more towards defining a meaningful and competent level at which to stop the verification process. In general, any computer code is designed to generate numbers by using a selected set of algorithms in some logical, predetermined manner. Initially, the computer code goes through a debug stage in which the programmer attempts to remove all coding errors and to establish that the model is working as required. Then the code is utilized by others for problem solving. Because of the extremely large number of logic paths possible in any modern computer program, the user inevitably discovers certain paths which produce non-physical or absurd answers. This section describes the procedure used in the assembly verification process for RETRAN, and why RETRAN can be reasonably expected to correctly generate numbers according to the prescribed logic, algorithms and formulas.

The RETRAN verification process was an extensive effort in attempting to assure that the computer program performs the correct calculation and processes the data and input as required. The first step in this activity was to assure that all individual models were adequately documented. This was accomplished by constructing a table which listed (1) the type of physical process represented, (2) the available RETRAN models of the process, (3) the reference used to confirm the models, and (4) a short description of the model. The original references were reviewed and compared with the RETRAN documentation to assure consistency.

The next step was to assure proper coding of each model. The four semi-independent methods of accomplishing this were:

(1) An independent visual check of the coding by someone other than the original programmer
(2) Specifying input (driving the model) to produce known results
(3) Editing all required information entering a model and hand calculating the output
(4) Comparing results with output from other codes.

A list of the RETRAN models versus these four verification methods was used to assure that each individual RETRAN model was verified against at least two of the four methods. Volume IV of CCM-5 gives the details of this effort.

Finally a series of "basic" test problems were run for which either analytical solutions were available or self-consistency checks could be made. Again the list of RETRAN models was cross-correlated against this list of problems to assure that all common mode models were correctly functioning. Some comparisons with experimental data, which are considered to be consistent applications, were included in this exercise. The results of this activity are also detailed in Volume IV.

TRANSIENTS OF INTEREST

Most applications of RETRAN to reactor analysis fall in the extended application category. This is because one or more of the basic assumptions used to develop the RETRAN models are in some way violated. The main assumptions which result in theoretical limitations can be classified as follows:

1. (1) One-dimensional assumptions
2. (2) Homogeneous flow assumptions
3. (3) Thermal equilibrium assumptions
4. (4) Steady-state correlations for most constitutive models, (e.g., heat transfer, critical flow, and friction factors).

Probably the most apparent limitation is that of one-dimensional streamline flow. Light water reactors (LWRs) have numerous regions where there are definitely multi-dimensional effects (e.g., the downcomer, steam generator, upper and lower plena, and the reactor core). However, if one is not interested in detailed distribution information in these regions, the multi-dimensionality may have only a minor influence on the bulk parameters of interest. Circumstances where this is most likely to be true are those cases in which only minor changes in system conditions occur and for those cases in which the change in a value, not the absolute value, is of interest.

It is thus necessary to categorize the various types of reactor analyses and to determine the code sensitivity and limitations in each case. Currently, RETRAN has been applied to most SAR Chapter 15 transients for BWRs and PWRs. Some of the normal operation and moderate frequency incidents produce mild transients in which the system variables are only slightly changed. For these cases, the initial conditions and the reactor control system can make a significant contribution to the plant response. Thus the steady-state and operational transient models in RETRAN are of great importance for these events. For other incidents and limiting fault events, the transients may produce large changes in the system conditions. In the case of a LOCA, these changes are usually the result of rapid depressurization, and/or ECCS injection, and may require use of refill/reflood models which are not in the RETRAN-01 version. RETRAN-01 has been applied only to the blowdown portion of primary pipe break transients.

For those normal and moderate frequency events which only slightly perturb reactor conditions, it has been demonstrated that, even though RETRAN is being applied in the extended application range, reasonable results can be obtained. The accuracy of the results has been determined by direct comparison with experimental data for those cases where such data exist. In this manner, confidence has been established in the calculation of other transients which are governed by similar phenomena. The details of this effort are the subject of the next section.

RETRAN QUALIFICATION

The qualification of a code is the comparison of the code results with experimental data so as to determine its applicability and sensitivity. The RETRAN Code has been qualified against three different classes of data, with the results summarized in Volume IV of EPRI CCM-5.

The first and simplest types of comparisons are separate effect experiments which, in general, are small scale tests where complexities are held to a minimum and the governing parameters accurately measured. The results of these analyses give confidence in both the individual, and combinations of, models utilized. Chapter III of Volume IV describes the qualification of RETRAN against the Separate Effect Tests.

The second category of experiments are system effects tests in which the interaction of various components must be described. In general these are intermediate size tests (e.g., Semiscale, Two-Loop Test Apparatus, LOFT) in which the assumption of one-dimensional streamline flow is reasonably approximated. The RETRAN qualification against experiments of this type is much more demanding and gives confidence in, and implies limitations on, the basic theory used in RETRAN. The results of these analyses are given in Chapter IV of Volume IV.

The third and most important class of comparisons is for plant data. The data from large nuclear plants are generally obtained from operational instrumentation and usually are limited in both quantity and quality. The actual comparisons are also made more difficult by the unavailability of all required input data (e.g., information regarding the initial conditions, time manual operator action was taken, response characteristics of valves). This phase of the RETRAN qualification was performed mainly by the EPRI/Utility System Analysis Working Group.

This group was primarily interested in performing analyses of their reactors, including some operational transients generally addressed in Chapter 15 of most Safety Analysis Reports. Each of the participating utilities identified both analyses of interest and some existing reactor start-up and/or operating tests against which to qualify the RETRAN results. Results of this effort are presented in Tables II and III, which list Chapter 15 incidents for BWRs and PWRs and show the transients analyzed and the degree of qualification of the RETRAN analysis which was possible.

The analyses are classified as:

(1) Direct comparison with experiments
(2) Results appear reasonable and agree with SAR results or calculations performed with other codes.
(3) Sensitivity studies performed.

The first category implies that the accuracy of the results was determined by direct comparison with experimental data for some (one or more) cases where such data exist. The second category is one where no experimental data were available and the results were reviewed only to assure that they appeared physically correct, and that they are consistent with results of other computer analysis, principally vendor SARs. Note however, that comparisons of RETRAN and SARs are of limited value unless the assumptions and models utilized by the vendor are known. In general RETRAN has to be used in a restricted manner to compare with these analyses. The third column shows the sensitivity studies documented as part of the Working Group Activity. These sensitivity studies, along with the additional analyses being performed with the code, contribute significantly to the qualification of RETRAN.

TABLE II

RETRAN ANALYSES OF PWR SAR-CHAPTER 15 TRANSIENTS

	Data Comparison	SAR-Other Calculations	Sensitivity Studies	COMMENTS
UNPLANNED DECREASE IN SECONDARY HEAT REMOVAL				
° Loss of external load	1	3	5	Startup Test
° Turbine trip				
° Loss of condenser vacuum				
° Steam pressure regulator failure				
° Loss of normal feedwater flow		2	6	
° Loss of A-C power to auxiliaries				
UNPLANNED INCREASE IN SECONDARY HEAT REMOVAL				
° Excessive load increase				
° Idle loop startup				
° Decrease in feedwater temperature		1	3	
° Increase in feedwater flow rate				
° Increase in steam flow rate				
° Inadvertent opening of steam generator relief or safety valve	1	2		TMI-2 Cooldown
CHANGES IN REACTOR COOLANT SYSTEM INVENTORY (PRIMARY SIDE INITIATED)				
° Inadvertent operation of ECCS				
° Accidental depressurization				
LOSS OF REACTOR COOLANT FLOW				
° Partial loss of flow	6	1	15	Pump Coastdown Test
° Complete loss of flow		6		
° Locked rotor				
REACTIVITY INSERTION (PRIMARY SIDE INITIATED)				
° Uncontrolled rod withdrawal				
- from subcritical		1	6	
- from power		6	14	
° Control rod misoperation				
° Chemical system malfunction				
ANTICIPATED TRANSIENTS WITHOUT SCRAM				
STEAM LINE BREAK		3		
RECIRCULATION LINE BREAK				
° Steam generator tube rupture				
° Small break				
° Loss of coolant accident		3		

TABLE III

RETRAN ANALYSES OF SAR-CHAPTER 15 BWR TRANSIENTS

	Data Comparison	SAR-Other Calculations	Sensitivity Studies	COMMENTS
PRESSURE INCREASE TRANSIENTS				
° Generator load rejection	2	4	2	Startup Test
° Turbine trip with/without bypass	2	6	20	Peach Bottom Tests
° Steam Line isolation valve(s) closure		1		
° Pressure regulator failure (close)				
° Loss of condenser vacuum				
° Turbine control valve fast closure				
° Above incidents with delayed scram				
REACTOR VESSEL WATER TEMPERATURE DECREASE				
° Loss of feedwater heater		5	9	
° Inadvertent injection pump start				
REACTOR VESSEL COOLANT INVENTORY DECREASE				
° Loss of feedwater flow	1	1		Startup Test
° Pressure regulator failure (open)				
° Relief or safety valve failure (open)				
° Loss of auxiliary power				
CORE COOLANT FLOW DECREASE				
° Recirculation pump seizure				
° Recirculation pump(s) trip	3	2	2	Startup Test
° Recirculation flow control failure				
CORE COOLANT FLOW INCREASE				
° Recirculation flow control failure				
° Startup of idle recirculation pump				
POSITIVE REACTIVITY INSERTION				
° Continuous rod withdrawal				
° - from subcritical				
- from power				
° Rod ejection				
ANTICIPATED TRANSIENT WITHOUT SCRAM				
STEAM LINE BREAK				
RECIRCULATION LINE RUPTURE				
° Large break				
° Small break				

COBRA-TF REFLOOD MODEL AND ANALYSIS OF FLECHT REFLOOD EXPERIMENTS

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ABSTRACT

COBRA-TF[a], a two-fluid, three-field transient thermal hydraulic analysis code is being developed at PNL. To simulate bottom reflood tests both a reflood entrainment model and a quench front model have been developed.

At the froth front, the entrained liquid fraction is determined by integration of the Nukiyama-Tanasawa droplet size distribution function up to a critical drop diameter. From this entrained fraction the entrainment rate is calculated and enters the drop number density, continuity and momentum equations as a source term. These conservation equations are solved in conjunction with the field equations for both vapor and continuous liquid to resolve the liquid carryover and vapor de-superheat.

A "fine mesh-rezoning" quench front model has been developed to overcome the numerical difficulties associated with simulating reflood heat transfer with large axial computational mesh spacings. Detailed representations of the surface heat flux and temperature profiles in the vicinity of the quench front are realized by overlaying a variable fine mesh upon the coarse hydrodynamic noding.

Qualification of this model is underway and several FLECHT forced reflood tests (3 - 15 cm/sec reflood rate) have been simulated. Generally good data comparisons were achieved, though areas for further improvement were indicated.

INTRODUCTION

The COBRA-TF computer code [1] is being developed as part of the NRC Water Reactor Safety Research Program in the area of analysis development. The purpose of this work is to provide better digital computer codes for computing the behavior of full-scale reactor systems under postulated accident conditions. The resulting codes are presently being used to perform pre- and post-test analysis of light water reactor component and system effects experiments.

COBRA-TF is formulated to model fully three-dimensional two-phase flow using a three-field representation. The fields are the vapor field, the

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[a] COolant Boiling in Rod Arrays - Three Field

continuous liquid field, and the droplet field. The model allows thermal nonequilibrium between the liquid and vapor phases and allows each of the three fields to move with different velocities. This paper describes the entrainment model and heat transfer methodology incorporated in COBRA-TF to enable its use as a calculational model for reflood/rewet simulations.

Entrainment of liquid droplets by steam is of extreme importance to thermal-hydraulic analyses of reactor LOCA's due to its effect upon pre-cooling of the fuel rods, the liquid inventory in the vessel, and steam binding from drops being carried over to the steam generator. To provide a mechanism for modeling entrainment, a third set of field equations has been added to COBRA-TF. The field equations for conservation of mass and momentum for the entrained droplet field follow.

MASS

$$\frac{\partial}{\partial t} \alpha_e \rho_\ell + \nabla \cdot (\alpha_e \rho_\ell \vec{U}_e) = -\Gamma_e''' + S''' \tag{1}$$

MOMENTUM

$$\frac{\partial}{\partial t} (\alpha_e \rho_\ell \vec{U}_e) + \nabla \cdot (\alpha_e \rho_\ell \vec{U}_e \vec{U}_e) = -\alpha_e \nabla P + \alpha_e \rho_\ell \vec{g} + \tau_{I_{ev}}''' - (\Gamma_e''' \vec{U}) + (S''' \vec{U}) \tag{2}$$

Constitutive equations to describe the rate processes of entrainment and de-entrainment, necessary as source terms for the field equations, are under development and will be incorporated in the code as they become available. The model currently used to determine the entrainment rate for reflood conditions is detailed below.

Coupled thermal-hydraulic numerical simulations of reflood encounter difficulties from the use of large axial computational mesh spacing (typically 1-2 feet) which cannot adequately resolve the axial profile of temperature and surface heat flux across the quench front. During quenching, the entire boiling curve -- from film boiling, through transition boiling and critical heat flux, to nucleate boiling -- can be encompassed by one computational cell. Constraining the entire cell to be in one boiling regime is nonphysical and results in stepwise cell-by-cell quenching, producing flow oscillations which can obscure the correct hydrodynamic solution. Consequently, an integration of the boiling curve through the hydrodynamic computational cell must be performed to determine the fluid heat input.

A "fine mesh-rezoning" technique [2] is employed in COBRA-TF to surmount these difficulties. Fine mesh heat transfer cells are superimposed upon the coarse hydrodynamic mesh spacing with axial and radial conduction and a boiling heat transfer package applied to each node. The reflood entrainment model and the "fine mesh-rezoning" quench front model are detailed below, along with data comparisons to a series of FLECHT forced reflood tests.

ENTRAINMENT MODEL

The reflood entrainment model is part of the "hot wall" flow regime used in COBRA-TF. Whenever the temperature of the heat transfer surface exceeds the critical heat flux temperature the "hot wall" flow regime is evoked. Figure 1 illustrates how this regime may look during bottom reflood. Below the quench front, single-phase liquid or bubbly flow is expected and the normal COBRA-TF flow regime logic is used to determine which set of models should be used for interfacial drag, interfacial heat transfer, etc. Above the quench front, the inverted annular flow regime is used. As shown in Figure 1, at high liquid volume fractions the liquid is assumed to consist of a continuous liquid column with vapor flowing around it. Liquid entrainment is assumed to be negligible in this regime until, at the froth front, the liquid column becomes unstable and shatters into droplets of various sizes.

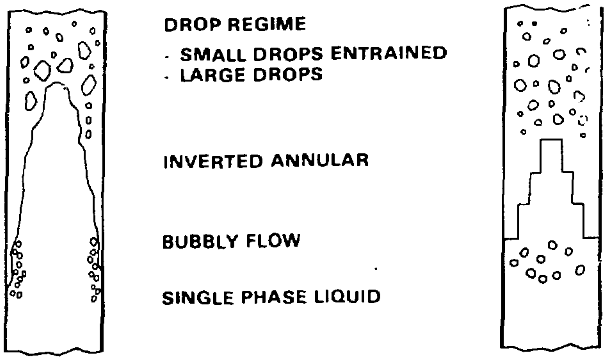


Figure 1. Schematic of the "Hot Wall" Flow Regime During Bottom Reflood

When the column shatters, a population of droplets is formed ranging from macroscopic (limited by channel dimensions) to microscopic in size. Of these drops, some will be carried away by the vapor flow (entrained) and the rest will fall back to the liquid column. The liquid in this region is treated as "large" drops for the liquid field and "small" drops for the entrained liquid field.

The maximum stable drop size in this entrainment region can be characterized by the critical Weber number,

$$D_o \doteq 13 \cdot \sigma/(\rho_v \cdot U_v^2) \tag{3}$$

The droplet size spectrum is determined from the Nukiyama-Tanasawa distribution function [3] in the form

$$P(D) = 32 \cdot \frac{D^2}{D_o^3} e^{-4D/D_o} \tag{4}$$

The maximum size droplet which can be entrained is given by the balance between buoyant and drag forces

$$D_{MAX} = \frac{3}{4} \cdot \frac{C_D \rho_v U_v^2}{(\rho_\ell - \rho_v) gc} \tag{5}$$

All drops smaller than D_{MAX} can be entrained and the entrained liquid fraction (η) is determined by integration of the droplet size spectrum multiplied by the drop volume over the region of entrainable drop diameter.

$$\eta = \frac{\pi}{6} \int_0^{D_{MAX}} D^3 \cdot P(D) \cdot dD \Big/ \frac{\pi}{6} \int_0^{\infty} D^3 \cdot P(D) \cdot dD \tag{6}$$

Equation (6) is used to obtain the source term for entrainment in the COBRA-TF difference equations. This term is simply

$$S''' = (1 - \alpha_v)\rho_\ell \frac{d\eta}{dt} - \eta\nabla \cdot \quad \alpha_\ell \rho_\ell \vec{U}_\ell \tag{7}$$

Once the drops are entrained, an average drop diameter must be used to compute the total interfacial drag and heat transfer. Since the volume to area ratio is the most important to maintain, the Sauter mean diameter is used.

Application of this model for the "hot wall" regime, when combined with the use of individual momentum equations for each of the three phases (drop, vapor and continuous liquid), allows the direct calculation of carryover and eliminates the diffusion of the continuous liquid phase.

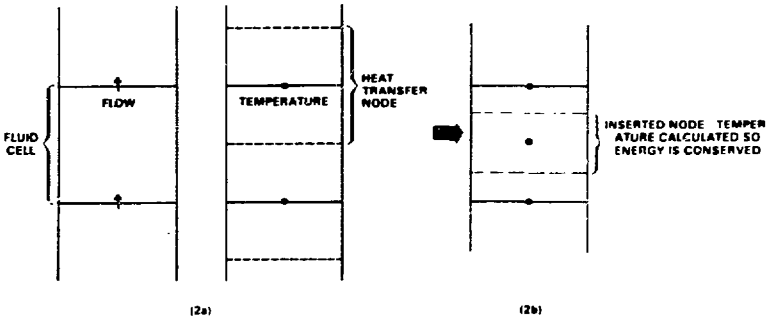
QUENCH FRONT MODEL

To extend the rod conduction model to resolve the severe thermal gradient associated with quenching, a fine mesh-rezoning technique was developed. By solving the two-dimensional conduction equation for a variable fine mesh at the quench front, propagation due to either quenching or dryout can be resolved and the surface heat flux integrated to provide the cell-averaged phasic heat inputs for the fluid energy equation. Also, the resulting quench front velocity will be a function of:

- axial conduction
- boiling curve shape
- prequench heat transfer
- pellet-clad heat transfer.

Resolution of axial temperature and surface heat flux excursions is achieved by rezoning the heat conductor mesh in their vicinity. Figure 2a illustrates the axial noding scheme normally employed by COBRA-TF. Both flows and rod temperatures are calculated at the boundaries of the fluid continuity cell.

When axial temperature differences exceed "splitting" criteria (maximum clad temperature differences specified by input) a row of nodes is inserted at the axial level halfway between the original nodes (Figure 2b). The temperatures assigned to these nodes are computed so energy is conserved. This splitting process continues (over a succession of time steps) until the mesh is fine enough.



(2a) (2b)

Figure 2. Example of COBRA-TF Noding Scheme and Fine Mesh Rezoning

The "fine mesh-rezoning" model is differentiated from other reflood models (e.g., that employed in RELAP-IV-MOD6(4)) in that the fine mesh nodes are stationary and do not have a fixed mesh spacing. Instead, the fine mesh nodes are split to create a graduated mesh spacing that readjusts itself

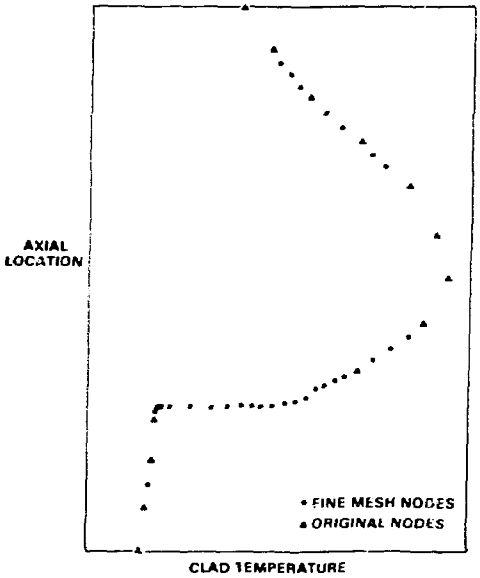


Figure 3. Axial Temperature Profile for FLECHT Test #00904 Showing Location of Fine Mesh Nodes

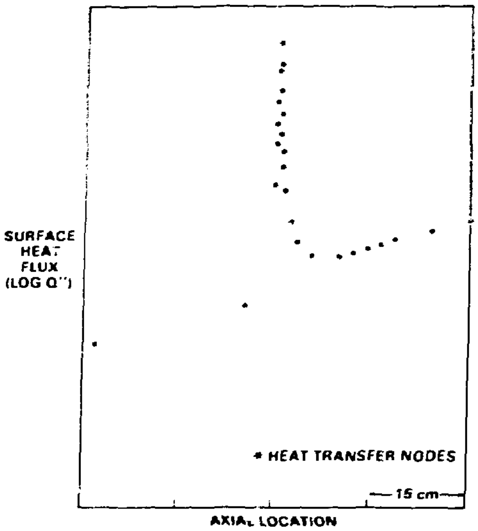


Figure 4. Axial Profile of the Surface Heat Flux at the Quench Front

consonant to the changing axial temperature gradient. This approach was implemented to enable the use of node sizes small enough (e.g., ∿ 0.05 cm) to resolve axial conduction and boiling curve shape at the quench front and yet minimize the number of nodes necessary, ensure conservation of stored energy, and simplify coupling with the hydrodynamic solution. Figure 3, taken from a simulation of a Westinghouse FLECHT low flooding rate test, illustrates this graduated mesh. Figure 4, a plot of surface heat flux versus axial position for the temperature profile of Figure 3, details the boiling curve shape at the quench front. Note: The hydrodynamic mesh spacing was 1 ft (30.5 cm) for this simulation.

EXPERIMENTAL DATA COMPARISONS

To assess the combined performance of the heat transfer, quench front and "hot wall" entrainment model, a series of FLECHT forced reflood experiments were wimulated. The operating conditions for these tests are given in Table 1. One channel, fourteen axial nodes and two heat transfer surfaces (heater rod and channel wall) were used in the COBRA-TF model of the FLECHT facility with a flowrate and enthalpy boundary condition applied at the beginning of the heated length. No

	Test		
	#3541	#4321	#00904
Initial Peak Clad Temperature (°K)	1143	1146	810
Peak Power (kW/m)	4.07	4.07	2.79
Flooding Rate (cm/sec)	15	9.3	3.76
Coolant Temperature (°K)	337	339	326
System Pressure (mPa)	0.393	0.40	0.283

Table 1. FLECHT Forced Reflood Run Conditions

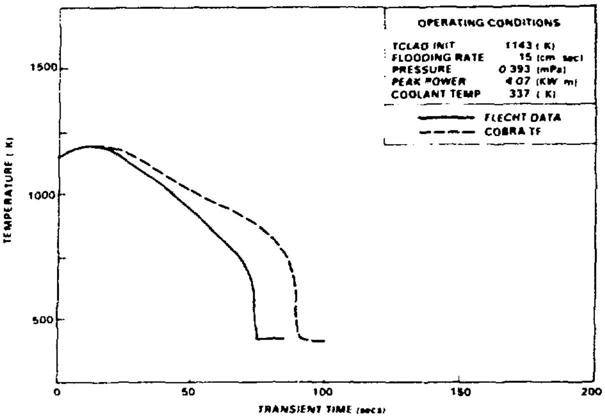


Figure 5. FLECHT Test #3541 - Hot Rod Midplane Clad Temperature History

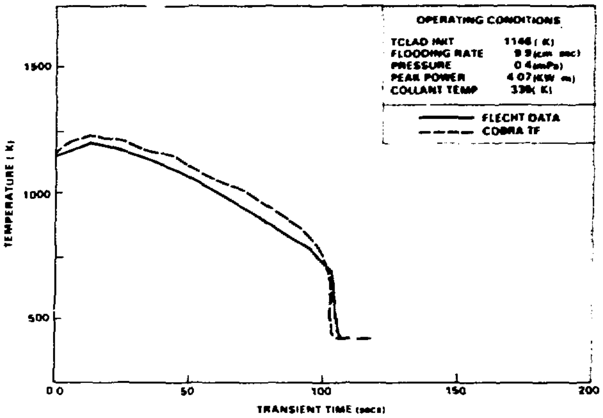


Figure 6. FLECHT Test #4321 - Hot Rod Midplane Clad Temperature History

attempt was made to model the details of the upper plenum and upper plenum de-entrainment.

Figure 5, clad temperature versus time for the hot rod, midplane, shows a comparison of the predicted and measured temperature histories for test #3541 [5]. Both turnaround time and peak temperature are predicted well, however, later in the transient the heat transfer is under-predicted resulting in higher clad temperatures and a later quench time. This discrepancy was not seen in the simulation of test #4321 and has since been determined to result from a discontinuity in the calculated critical heat flux as a function of mass flowrate. The mass flowrate in test #3541 was such that near the quench front the mass flux had approximately the same value as the logic switch between the two equations of the Biasi CHF correlation. Consequently, switching between equations resulted in oscillations that caused too much carryover resulting in less liquid inventory and smaller precooling due to inverted annular film boiling.

Test #4321 [5] is illustrated in Figure 6. Comparison between calculation and experiment is very good for the entire transient. As noted above, due to a lower flooding rate than test #3541, the CHF correlation switching oscillations were not present in this simulation.

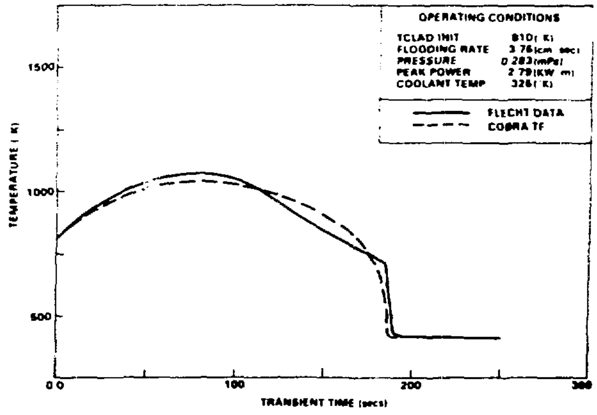


Figure 7. FLECHT Test #00904 - Hot Rod Midplane Clad Temperature History

Test #00904 [6], shown in Figure 7, is a low flooding rate test (3.76 cm/sec) and is differentiated from tests #3541 and #4321 in that the major mode of precursory heat transfer is dispersed flow film boiling rather than inverted annular film boiling. During the first 120 seconds of the transient the rod temperature history is predicted well, under-predicting the peak clad temperature by less than 20°C after a heatup of more than 250°C. Typical computed values of the quality and void fraction during the period were 40-50% and 0.944-0.996 respectively.

Between 120 and 160 seconds the wall heat transfer is underpredicted due to either an overprediction of the entrainment rate or too little interfacial heat transfer between the superheated vapor and the droplets, or a combination of both. However, the froth level reaches the midplane at approximately the correct time, inverted annular film boiling is established at ∿180 seconds, and the surface temperature falls rapidly to the quench temperature.

The good comparisons between measured and predicted void fractions for run #00904, Figure 8, illustrate the ability of the "hot wall" entrainment model, coupled with the three-field formulation, to model reflood entrainment. The experimental void fractions were determined from the output of pressure difference (DP) cells located every two feet along the bundle. As such, these void fractions are an indicator of the liquid mass inventory in the bundle and are especially important in low reflood rate tests. For example, in test #00904, approximately 25% of the inlet mass flow is boiled off, another 60% is carried over as entrained liquid droplets, leaving just 15% of the inlet flow to contribute to the liquid mass inventory. Consequently, a small error in the entrainment can, when integrated over time, lead to a large discrepancy in the mass inventory and result in predicted quench times much different than the experiment.

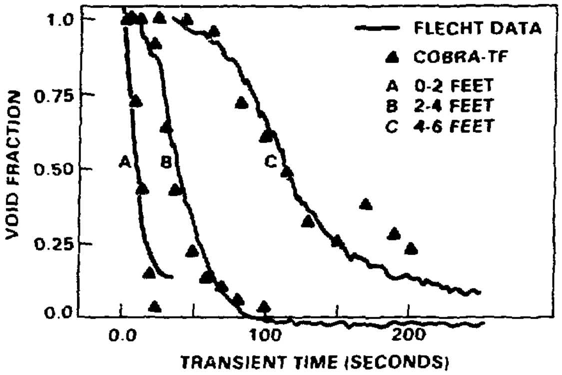


Figure 8. FLECHT Test #00904 - Comparison of Predicted and Measured Void Fractions at Three Axial Locations

An example of the enhanced entrainment modeling capability of the three-field model is shown in Table 2, a copy of code output for test #00904. The treatment of the fluid in the froth region as a two group drop model (large and small droplets) results in the maintenance of a sharp liquid level interface virtually eliminating numerical diffusion and predicting void fractions and droplet sizes (0.966 and 0.020 cm, respectively) in the range of those observed in the FLECHT movies.

```
Simulation Time:  50.0 seconds

                                   Velocity
Node    Dist.     Pressure         (ft/sec)                  Volume Fraction
No.     (ft)       (psi)     Liquid   Vapor    Entr.     Liquid    Vapor    Entr.
 14     13.00     41.014      0.00    66.28    31.18     .0000    .9968    .0032
 13     12.00     41.051      0.00    64.76    33.31     .0000    .9969    .0031
 12     11.00     41.086      0.00    63.93    32.11     .0000    .9967    .0033
 11     10.00     41.108      0.00    62.73    30.78     .0000    .9964    .0036
 10      9.00     41.142      0.00    61.64    29.44     .0000    .9962    .0038
  9      8.00     41.166      0.00    58.59    28.14     .0000    .9959    .0041
  8      7.00     41.196      0.00    54.40    27.22     .0000    .9957    .0043
  7      6.00     41.224      0.00    46.39    26.64     .0000    .9955    .0045
  6      5.00     41.212      0.00    38.55    26.74     .0552    .9402    .0045
  5      4.00     41.443      0.00    79.27    29.59     .6783    .3182    .0035
  4      3.00     41.833       .12    38.98    39.64     .9870    .0130    .0000
  3      2.00     42.244       .13      .13      .13     .9979    .0021    .0000
  2      1.00     42.666       .12      .12      .12    1.0000    .0000    .0000
  1      0.00     41.010       .12      .00      .12    1.0000    .0000    .0000
```

Table 2. COBRA-TF Output for Reflood Test #00904 at 50 Seconds Transient Time

SUMMARY

Reflood simulation capability has been incorporated in the COBRA-TF code through the inclusion of a third set of field equations for the entrained droplet field, and development of the "hot wall" flow regime and the "fine mesh-rezoning" quench front model. Simulations of three FLECHT bottom reflood tests were employed to assess the integrated performance of the quench front, heat transfer, entrainment and hydrodynamic models of COBRA-TF. Generally good data comparisons were achieved though need for improvements in the heat transfer and entrainment models were indicated. Further testing is needed and a series of simulations using the more recent FLECHT skewed profile reflood tests will be used to compare measured and predicted clad temperatures, void fractions and vapor superheat.

NOMENCLATURE

C_D = droplet drag coefficient

D = drop diameter (m)

$\vec{g}$ = gravitational acceleration (m/sec^2)

P = pressure (N/m^2)

S''' = entrainment rate (kg/sec $\cdot$ m^3)

t = time (secs)

U = velocity

α = phase volume fraction

Γ''' = vapor generation rate (kg/sec m^3)

η = entrained fraction

σ = surface tension (N/m)

τ_I''' = interfacial shear (N/m^3)

Subscripts

e = entrained phase

ℓ = liquid phase

v = vapor phase

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THE BOILING CURVE DURING REFLOODING

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ABSTRACT

Studies of heat transfer during the reflooding phase of the LOCA were done using experimental measurements from the separate effects test facility EVA/KWU. The transient behaviour of the surface heat flux deduced from measured wall temperature was found to include most of the heat transfer regimes defined in the boiling curve. This implies that the cooling process during reflooding follows the boiling curve from high to low temperatures.

The thermal hydraulic code REFLUX/GRS which includes a fairly good simulation of the most important processes occuring during reflooding is dicussed. This code was used to predict the behaviour of different key parameters such as : wall temperature, pressure drop, quench front and mass flow rate of entrained water for different test facilities. Agreement was found between measured and predicted values. Pre- and post-test calculations of International Standard Problem No.7 showed the advantages of REFLUX/GRS and also its weak points.

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INTRODUCTION

Theoretical and experimental activities in different countries are going on to better understand the phenomena accompanying the reflooding phase of a LOCA in light water reactors. One important phenomena is the delivery of ECC water to the core region. Reflooding of the core is then dependent on the heat transfer mechanism taking place between the fuel Pin cladding and the fluid present. Separate effect tests to study the heat transfer mechanism during reflooding were carried out or are underway in different heat transfer laboratories. The test section is generally composed of electrically heated monotubes or rod bundles.

HEAT TRANSFER MECHANISM DURING REFLOODING

From the measurements on a well instrumented heated tube (EVA experimental facility /1/ of KWU) the time dependent surface heat flux during reflooding was deduced. All experiments analysed /2/ show a distinct maximum in the heat flux. This point is defined to be the rewetting time of the given position. A plot of the quench time against the axial position gives enough information to determine the propagation of the quench front. To find out the sequence of heat transfer regimes appearing at a certain axial position, the calculated heat flux is plotted against the surface wall temperature. At the central region of the test section one can differentiate between 5 different heat transfer regimes, which can be identified as dispersed flow film boiling, inverted annular film boiling, transition boiling, nucleate boiling and forced convection to liquid (Fig. 1). This means that cooling process during reflooding is a sequence of the heat transfer regimes given in the boiling curve.

A THERMAL HYDRAULIC CODE TO PREDICT CLADDING TEMPERATURE BEHAVIOUR

In order to predict the temperature behaviour of the hot wall (tube or cladding) during reflooding a computer code is needed that considers not only the different heat transfer regimes but also all other physical phenomena observed. The computer code REFLUX/GRS developed for this work includes a fairly good simulation of the most important processes occuring during reflooding. This is demonstrated through the good prediction of the OECD-CSNI Standard Problem No.7 /3/ (ERSEC reflooding experiment) shown in Figs. 4-9

The solution logic of REFLUX/GRS is based on an effective coupling of the three main parts described below:

1. A heat conduction and heat generation model which is one dimensional in radial direction and uses temperature dependent material properties. Based on these assumptions models to simulate nuclear fuel rods, electrically heated pins and tubes directly heated by Joule effect are included. The influence of axial heat conduction on the rewetting process is considered through a quench front propagation model.

2. A surface heat transfer model which includes heat transfer correlations for 9 different heat transfer regimes is adapted. Experimentally verified correlations to determine the three important points on the boiling curve (temperatures of minimum film boiling, rewetting and critical heat flux). The transition from one heat transfer regime to an other depends mainly on:
   - The value of the calculated surface temperature ϑ_w with respect to the minimum film boiling temperature ϑ_{min}, the rewetting temperature ϑ_o and the critical heat flux point ϑ_{CHF}
   - The axial position of the swell level
   - Teh axial position of the quench front
   - The value of the superficial liquid velocity J_f
   - The value of the superficial steam velocity J_g with respect to the critical value J_{gc}.
   - The value of the fluid temperature with respect to the saturation temperature
   - The droplet diameter.

   The heat transfer correlations used are given in table 1.

3. The fluid dynamic model is based on the MIT code REFLUX /4/. The most important modfications introduced are:
   - The finite difference equations were modified to account for the dependence of the fluid density on local conditions.
   - The dependence of thermodynamic properties on local pressure and temperature is recognized by using a set of equations given in the international steam tables.

- The one dimensional drift-flux model described by Ishii is used for the flow regimes with a continuous liquid phase.
- The swell level is determined using a bubble rise model.
- Dispersed flow regimes are defined using the carry-over criterion from Plummer with a value of 1.5 for the critical Weber number.
- Droplet break up is considered using the assumptions given by Groeneveld, until a minimum droplet diameter of 1 mm is reached.

VERIFICATION OF REFLUX/GRS

Code verification was performed using selected tests from three different facilities with forced feed flooding.

1. EVA /1/ is a monotube test facility. The test section is 3m long and 11.8 mm internal diameter. The experiments were performed under constant boundary conditions of system pressure, initial wall temperature, power generation, flooding rate and inlet water subcooling. REFLUX/GRS was applied to predict the behaviour of the wall temperature during the flooding process. This was done for nine different tests. The agreement between measured and predicted values was very good in the regions where the influence of counter current flow due to an upper quenchfront was negligible as shown in fig.2 for test 109. In this test the initial wall temperature was 870 K, the flooding rate and the water temperature at the inlet were kept constant at 0.1 m/sec and 300 K respectively.

2. FLECHT experiments were predicted by W. Kirchner and P. Griffith using REFLUX /4/. As indicated in their work the worst prediction was obtained for RUN # 0284 (Initial Peak Temperature = 1139 K, Flooding Rate = 0.025 m/sec, Pressure = 0.145 MPa, Peak Power = 4.07 kW/m and Coolant Inlet = 357 K). Predictions of this Run using REFLUX/GRS showed better agreement between measured and calculated temperature behaviour as given in Fig. 3. This was due to the modifications introduced to the fluid dynamic Model as described above.

3. ERSEC is a monotube test facility at the nuclear research center in Grenoble (France). The test section consisted of an Iconel tube, uniformly heated by Joule effect and internally cooled. The boundary conditions of the CSNI Standard Problem No. 7 /5/ defined by French CEA were constant values of Pressure (0.3 MPa), flooding rate (5,2 g/cm^{2}s), inlet water subcooling (20°C) and power generation (5W/cm^2). The initial wall temperature of 600°C was uniform. A comparison between experimental data, pretest calculations and post test calculations using REFLUX/GRS are given in Figures 4-9. The only difference between pretest and post test calculations was the axial location of the wall temperature used in the quenchmodel. The oscillatory behaviour of the calculated mass flow rate of entrained water during the 1st 200 seconds shows that some modification of the entrainment model is necessary. The entrainment model adapted in REFLUX/GRS is based on the Plummer carry over criterion /6/ which assumes that water droplets can be carried by vapour only when the super-

ficial velocity of the vapour is equal or greater than a critical value. At that axial position the entire amount of liquid present is then assumed to be carried by the vapour. This abrupt change at the carry over point causes the oscillatory behaviour shown in Fig. 8. A gradual increase of the entrained liquid with the increase of the superficial velocity until the critical value is reached, will diminish this oscillatory behaviour.

This modification will cause an increase of entrained water in the upper region of the test tube, followed by higher heat flux from vapour to droplets which will decrease the vapour temperature. This implies that the overprediction of the vapour temperature at the outlet of the test section (Fig. 9) is also due to the entrainment model used.

CONCLUSIONS

The results discussed above indicate that the thermal hydraulic code REFLUX/GRS has the capability to predict heater rod surface temperatures during forced feed reflood fairly good. The quench model introduced using a one-dimensional (axial) conduction model is adequate for predicting the correct quench time at the different axial locations.

Modifications are required to improve the entrainment model. This requires more experimental information on the effect of various parameters on entrainment especially in the inverted annular flow regime.

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TABLE 1: HEAT TRANSFER CORRELATIONS

Heat Transfer Regime	Correlation used	Remarks	Reference
Forced Convection to vapour	$Nu = 0.023\, Re_D^{0.8}\, Pr_D^{0.3} \left\{1.0 + 0.3\left(\frac{D_H}{x_Q+0.01 D_H}\right)^{0.7}\right\} \left(\eta_D / \eta_l\right)^{0.14}$ (1)	x_Q = local distance from the quench front	Groeneveld
Dispersed Flow Film Boiling	q_{wD} is determined using eq. (1) $q_{DT} = \frac{\lambda_D}{\delta}\left[2+0.74 Re_\delta^{0.5} Pr^{0.33}\right] (\vartheta_D - \vartheta_{sat})$ (2)	Two step heat transfer is used: q_{wD} = wall to vapor and then q_{DT} = vapor to drop	Lee and Ryley
Inverted Annular Film Boiling	$h_{IAFB} = 0.62\left[\frac{g \cdot \rho_D(\rho_l - \rho_D) H'_{lD} \lambda_D^3}{\eta_D\, \Delta\vartheta\, D_H}\right]^{1/4} \left[\frac{D_H}{2\pi\sqrt{\sigma / g(\rho_l - \rho_D)}}\right]^{0.172}$ (3) $h_r = \frac{\sigma}{\frac{1}{\varepsilon}+\frac{1}{a}-1} \cdot \frac{T_W^4 - T_{sat}^4}{\Delta\vartheta}$ (4) $h = h_{IAFB} + 0.75\, h_r$ (5)	H_{lD} = modified latend heat of vaporisation $H'_{lD} = H_{lD} + \left[1 + \frac{0.5 C_{PD} \Delta\vartheta}{H_{lD}}\right]$	Bromley Pomeranz
Transition Boiling	$q_{CHF} = 0.15\, \rho_D H_{lD}\left[\frac{\sigma g(\rho_l - \rho_D}{\rho_D^2}\right]^{0.25}$ (6) $q_{min} = h_{IAFB}(\vartheta_{min} - \vartheta_{sat})$ (7)	Log-Log interpolation between q_{CHF} and q_{min}	Zuber
Nucleate Boiling	$q_{NB} = h_{NB}(\vartheta_w - \vartheta_{sat}) + h_{Fc}(\vartheta_w - \vartheta_l)$ (8) $h_{NB} = 0.00122\left(\frac{\lambda_l^{0.79}\, CP_l^{0.45}\, \rho_l^{0.48}}{\sigma^{0.5}\, \eta_l^{0.29}\, H_{lD}^{0.24}\, \rho_D^{0.24}}\right) \Delta T^{0.24}\, \Delta p^{0.75}\, S$ (9) $h_{Fc} = 0.023\, \frac{\lambda_l}{D_H}\, Re_l^{0.8}\, Pr_l^{0.4} F$ (10)	S = Suppression factor given by Chen F = factor for two phase flow	Chen
Forced Convection to single phase liquid	$Nu = \frac{h D_H}{\lambda_l} = 0.17\, Re_l^{0.33}\, Pr_l^{0.43}\, Gr_l^{0.1}$ (11)	$Re_l < 2000$	Collier
	$Nu = 0.0248\, Re_l^{0.8}\, Pr_l^{0.4}$ (12)	$Re_l > 2000$	Dittus Boelter

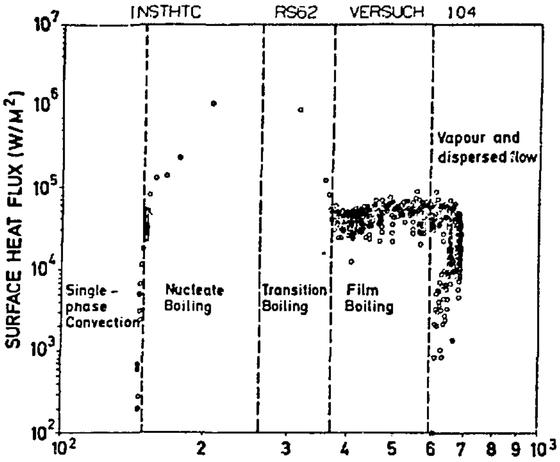

FIG. 1: VARIATION OF SURFACE HEAT FLUX WITH SURFACE TEMPERATURE AT THE CENTRAL REGION OF EVA

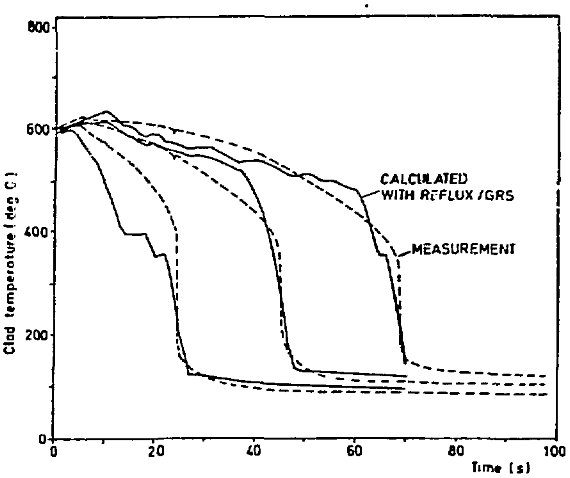

FIG.2
TEMPERATURE BEHAVIOUR AT THE ELEVATIONS 0.75,1.5 AND 2.3 M (EVA TEST 109)

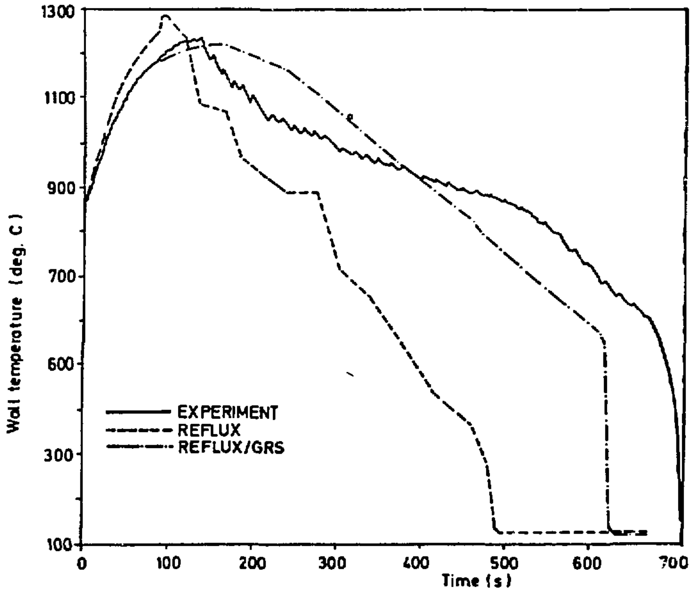

FIG. 3 FLECHT RUN # 0284 CLAD TEMPERATURE VS. TIME AT 6 FT. ELEVATION OF PIN 5F

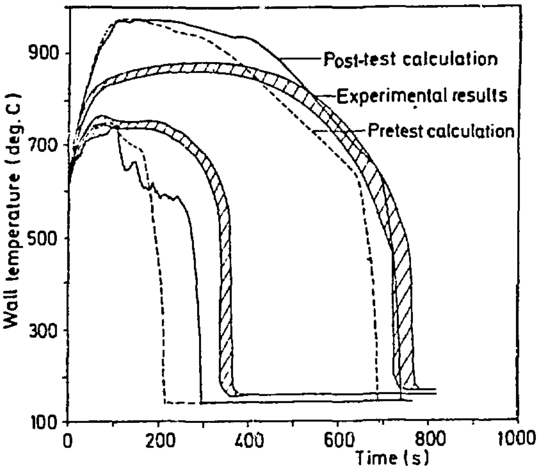

FIG.4 WALL TEMPERATURE AT ELEVATIONS OF 1.30 M AND 2.99 M

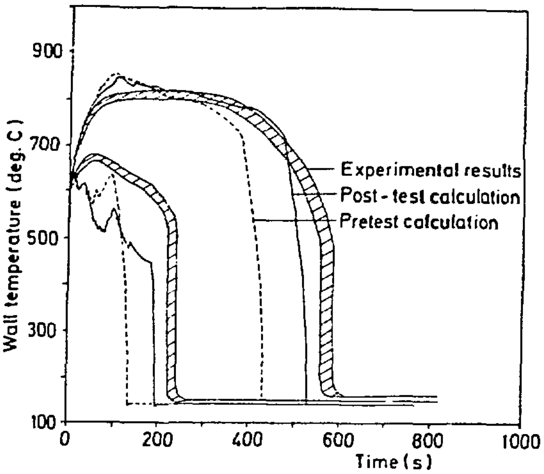

FIG. 5 WALL TEMPERATURE AT ELEVATIONS OF 0.98 M AND 2.03 M

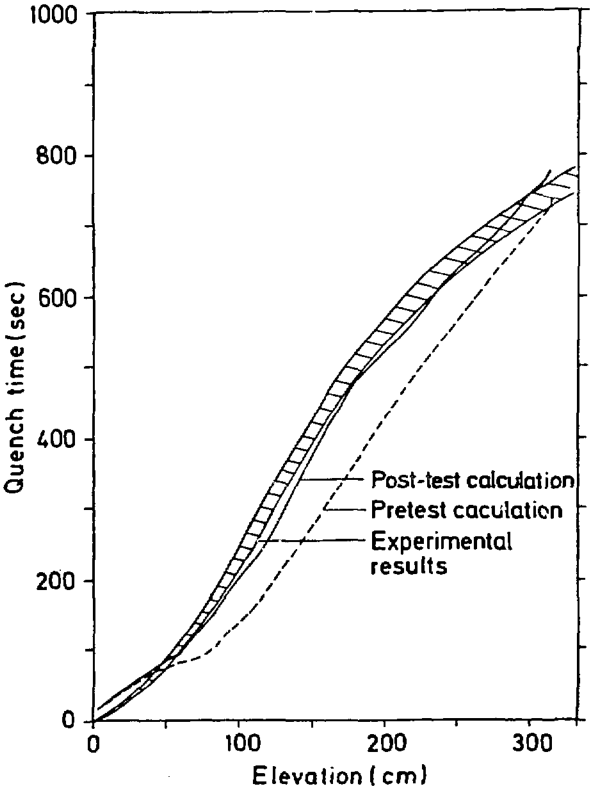

FIG. 6 MOTION OF THE QUENCH FRONT

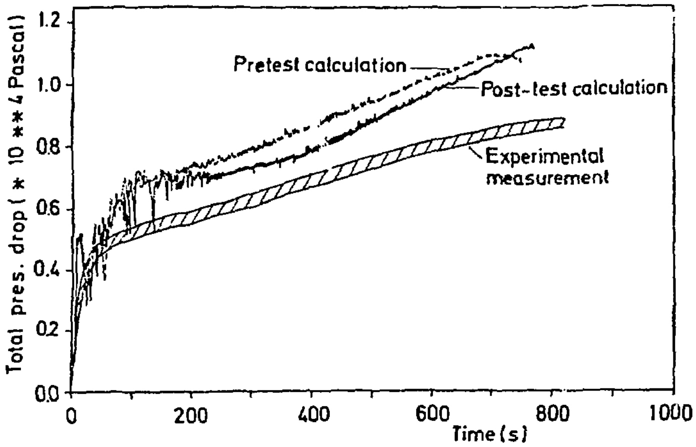

FIG.7 TEST SECTION TOTAL PRESSURE DROP

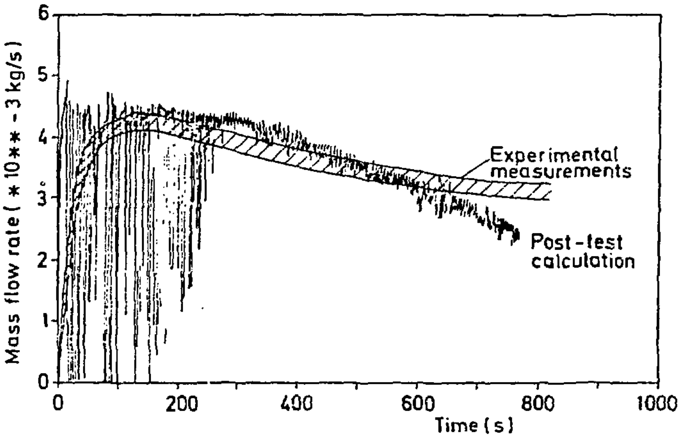

FIG.8 MASS FLOW RATE OF ENTRAINED WATER

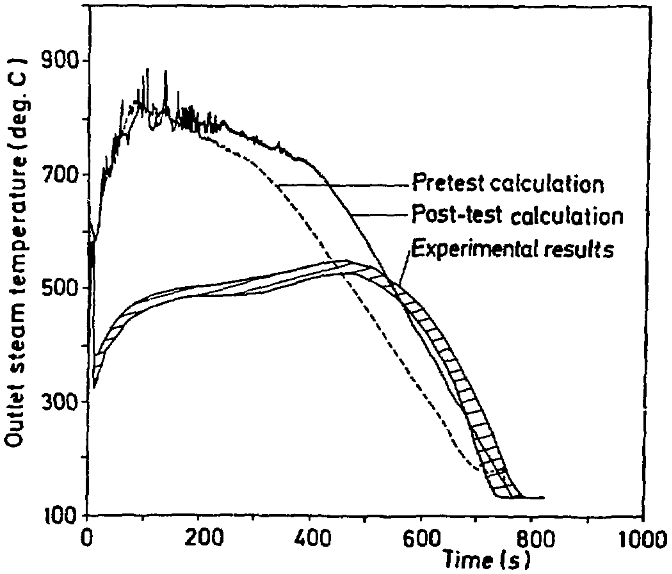

FIG.9 STEAM TEMPERATURE AT THE OUTLET OF TEST SECTION

Dup

LWR FUEL ROD BEHAVIOR OBSERVED DURING POSTULATED ACCIDENT CONDITIONS: A COMPARISON OF FRAP-T CALCULATED AND PBF EXPERIMENTAL RESULTS [a]

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ABSTRACT

Light water reactor (LWR) fuel rod behavior during transient experiments conducted in the Power Burst Facility is reviewed. The experiments examined simulated hypothetical reactivity initiated accidents (RIA) and power-cooling-mismatch (PCM) events. Fuel rod behavior calculated by the Fuel Rod Analysis Program-Transient (FRAP-T) is compared with the test data. Important physical phenomena observed during the tests and not presently incorporated into the FRAP-T code are: (a) fuel swelling in the radial direction due to fission gas effects, (b) UO_2-zircaloy chemical interaction, and (c) loss of UO_2 grain boundary strength and fuel powdering. Additional models needed in FRAP-T to reflect the fuel behavior observed during the two types of transients are cladding thickness variation during an RIA, molten fuel movement and possible cladding-molten fuel thermal interaction during a PCM event, and in the case of breached rods, the effects of hydrogen pickup on cladding embrittlement.

INTRODUCTION

LWR fuel rod behavior during various hypothetical off-normal and accident conditions is being studied in the Power Burst Facility (PBF). These irradiation experiments are part of the Nuclear Regulatory Commissions (NRC) Safety Research Program and emphasize the physical phenomena, failure thresholds, and damage mechanisms which occur during selected design basis accidents. The performance of the Fuel Rod Analysis Program - Transient (FRAP-T) which calculates thermal, mechanical, and chemical interaction behavior of fuel rods during off-normal and accident conditions in an LWR is evaluated based on the test results. The fuel behavior phenomena and the code calculations, as applied to various transients, are compared and assessed here from an experimenter's point of view. The purpose of the review is to identify possible modifications and additions to the models in the current computer code to better reflect the experimental evidence. Two types of accident conditions are reviewed:
(a) reactivity initiated accidents (RIA), and
(b) power-cooling-mismatch (PCM) events.

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[a] Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.

Reactivity Initiated Accidents

LWR type fuel rods were subjected to peak fuel enthalpies of 185 to 285 cal/g during the PBF RIA Test Series starting at hot-startup conditions typical of a commercial boiling water reactor. The failure threshold of previously irradiated fuel rods, up to a burnup of 4.6 GWd/t, was identified to be approximately 140 cal/g UO_2. The failure mechanism was brittle rupture or tearing of the cladding due to deformations at high strain rates caused by pellet-cladding mechanical interaction. Departure from nucleate boiling has no influence on this damage mechanism. The failure threshold of previously unirradiated rods was found to be between 225 and 250 cal/g UO_2, with the failure mechanism being brittle fracture of oxidized and deformed cladding. Flow blockage was observed at a fuel enthalpy of 285 cal/g UO_2. This enthalpy is similar to the NRC guideline (280 cal/g UO_2) for loss of coolable geometry.

At high enthalpies, the rods failed with severe cracking and crumbling of fuel and embrittled cladding. Significant wall thickness variations were observed in the cladding of fuel rods subjected to 225 to 285 cal/g UO_2. Regions of cladding thinning and thickening from about 60 to 170%, respectively (Figure 1), of original thicknesses were observed.[1] Such gross wall thickening and thinning was associated with partial or total melting of cladding at enthalpies of 225 cal/g UO_2 and above[2] and was apparently assisted by variations in the local coolant pressure associated with the rapid heating of the coolant during the power burst due to neutron and gamma heating of the coolant. Uniform oxidized layers of oxygen stabilized alpha zircaloy and zirconium oxide, developed around the circumference of both the inside and outside surfaces of the cladding. The oxidized inner layer resulted from the fuel-zircaloy reaction, and the outer layer from the zircaloy-water reaction. The thinned regions of the cladding were often fully oxidized. The oxidation rate was noted to increase significantly for temperatures above 1800 K.

The fuel in the unirradiated rods experienced fuel grain boundary separation and consequent fuel powdering due to the thermal stresses caused during quenching from the film boiling operation.[3] A considerable amount of the fuel powder washed out upon fuel fracture. The rods with burnup of 4.6 GWd/b experienced fission product induced swelling of molten or nearly molten fuel which resulted in cladding rupture and complete blockage of coolant flow shrouds at enthalpies of about 285 cal/g UO_2 as shown in Figure 2.

Fuel temperatures in the RIA tests were generally well calculated by the FRAP-T code;[a] however, the code undercalculated the severity of the mechanical deformation and breakup of the fuel rods at high energy depositions.

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[a] FRAP-T, MOD-004, Version 5/2/78, EG&G Idaho, Inc., Configuration Control Number H00372IB.

The important phenomena observed during RIA experiments in the PBF that are not presently modeled in FRAP-T include:

1. Fuel swelling due to fission gas and volatile fission product coalescence, release, and expansion (which strongly influences rod diametral deformation and the potential for coolant flow blockage);
2. Cladding thickness variation and the UO_2-zircaloy chemical interaction and oxidation (which together lower the failure threshold);
3. More realistic zircaloy-water reaction kinetics at higher temperatures up to 2100 K (the oxidation rate at temperatures above 1800 K is believed to be much higher than calculated by the existing Cathcart model[4]);and
4. Loss of UO_2 grain boundary strength (needed to calculate the amount of powdered fuel available for flow blockage).

Power-Cooling-Mismatch Event

The PBF tests simulating PCM events showed that fuel rods can operate in a film boiling condition and incur considerable damage without failure. The severity of the rod damage is primarily dependent on the power level, cladding temperature, and duration of high temperature operation. At temperatures above 920 K, cladding damage included cladding collapse and waisting.[5] For long periods of film boiling with cladding temperatures above 1200 K, oxygen embrittlement was observed from both the cladding-water and cladding-fuel chemical reactions.[6] The PBF in-pile test rod failures support Pawel's failure criteria.[7] However, additional room temperature cladding embrittlement, from a combined effect of hydrogen and oxygen, occurred with cladding hydrogen concentrations as low as 300 ppm in the prior beta material.[8] Hydriding contributed to embrittlement only in rods that had failed prior to or during film boiling and was probably the result of the presence of stagnant steam conditions inside the fuel rod.

Fuel damage included fuel swelling, molten fuel relocation, and grain boundary separation. Modest fuel swelling was observed in previously unirradiated rods due to thermal effects. However, fuel swelling occurred in preirradiated rods to a somewhat larger extent due to the additional effects of retained fission gas. Fuel swelling neither resulted in rod failure nor significantly affected the behavior of rods with burnups ranging up to 17,000 MWd/t during a PCM test. Fuel restructuring occurred within the film boiling zone after the formation of a central void within the high density, molten fuel (Figure 3). Equiaxed grain growth was observed around the region of previously molten fuel, with the grain size decreasing toward the pellet exterior. Fuel grain boundary separation (powdering) during quenching from film boiling operation occurred in both fresh and previously irradiated fuel.

Molten fuel contact with the cladding as a result of molten fuel relocation, with the potential for cladding melting, was observed. However, cladding melting did not occur in the few PBF tests in which molten fuel contacted the cladding.

Peak cladding temperatures calculated by FRAP-T matched the posttest estimated values very well (Figure 4) for the periods of film boiling operation, but the calculated fuel centerline temperatures were somewhat higher. The calculated temperature drops across the cladding at high power levels were much smaller than those observed during film boiling. The calculated rod elongation during film boiling followed the measured values relatively well (although 18% lower) as a function of time for rods that did not balloon. To more completely simulate the behavior of an LWR fuel rod during a PCM event, inclusion of the following phenomena, not currently modeled in FRAP-T, should be considered.

1. Fuel swelling in irradiated rods due to fission gas bubbles trapped at grain boundaries and in molten fuel (needed for calculation of diametral expansion of high burnup fuels during film boiling operation);
2. Central void formation during film boiling (needed to better calculate fuel temperature distributions);
3. Molten fuel movement, fuel freezing, and cladding thermal interaction (necessary to estimate the potential for cladding melting upon molten fuel contact);
4. UO_2-zircaloy chemical interaction and oxygen diffusion (necessary to calculate the degree to which it increases cladding embrittlement);
5. Loss of UO_2 grain boundary strength and fuel powdering (needed to calculate the potential for such phenomena to cause coolant flow blockages); and
6. Effects of hydrogen pickup on cladding embrittlement in the case of defective or breached rods (needed to calculate the failure thresholds for such rods).

On the basis of the observed phenomena, the following modifications to the existing FRAP-T models are also suggested. Pawel's room-temperature cladding embrittlement criteria, or the Chung and Kassner criteria[9] based on oxygen content in the beta zircaloy, should be adopted to properly account for posttest handling fractures. The increased degradation of cladding conductivity with increased oxidation should be modeled to better reflect the observed larger temperature drops across the cladding.[10]

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NOTICE

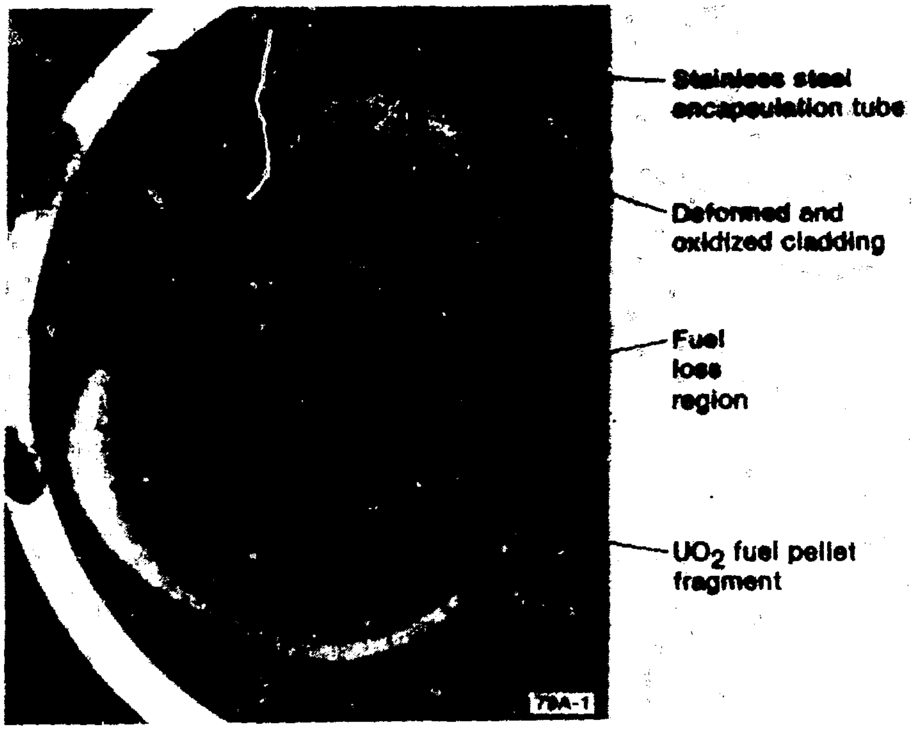

Figure 1. Test RIA-ST-1 fuel rod cladding near the peak flux location (0.35 m from the bottom of the rod).

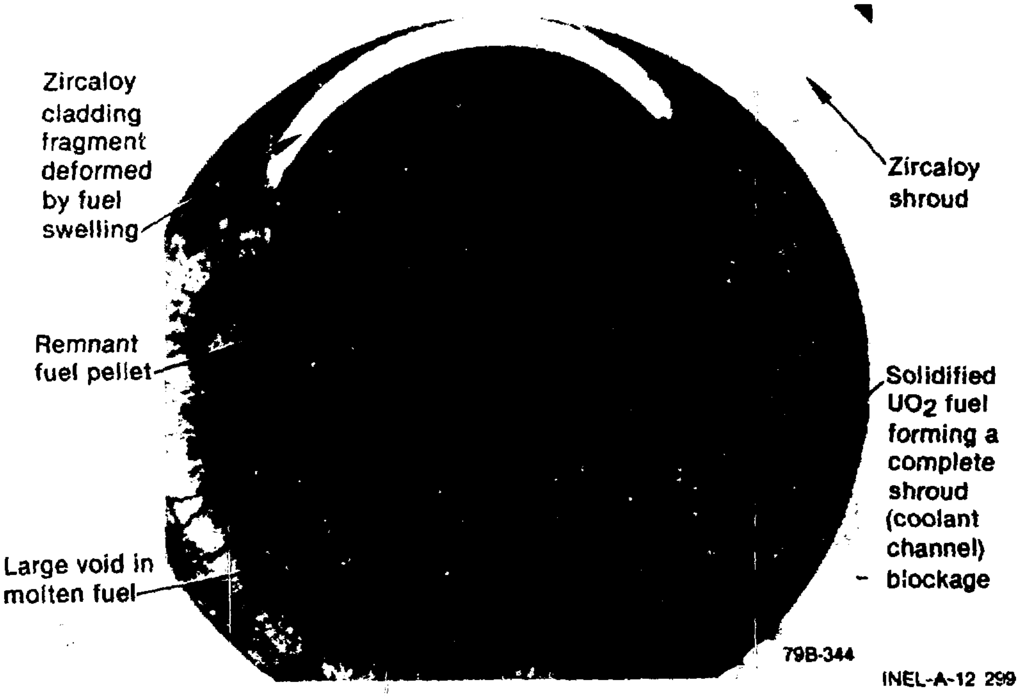

Figure 2. Ground and polished cross section of Rod 801-1 showing fuel, cladding, and shroud near the peak power elevation (Test RIA 1-1).

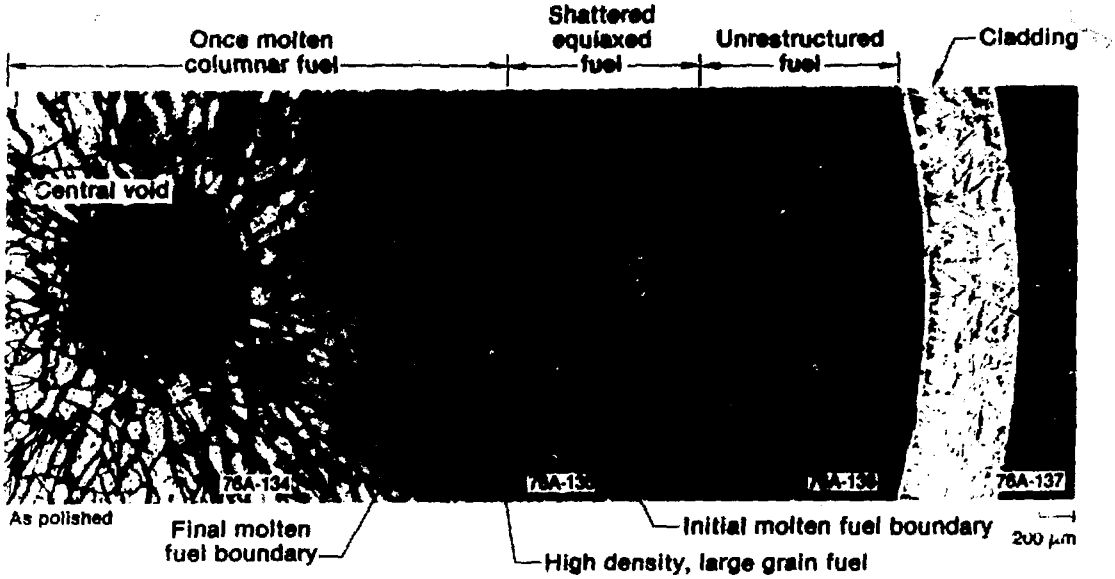

Figure 3. Transverse section showing typical fuel microstructure across a fuel pellet from the film-boiling zone. Sample from the unirradiated fuel Rod IE-001, following the IE scoping Test 1.

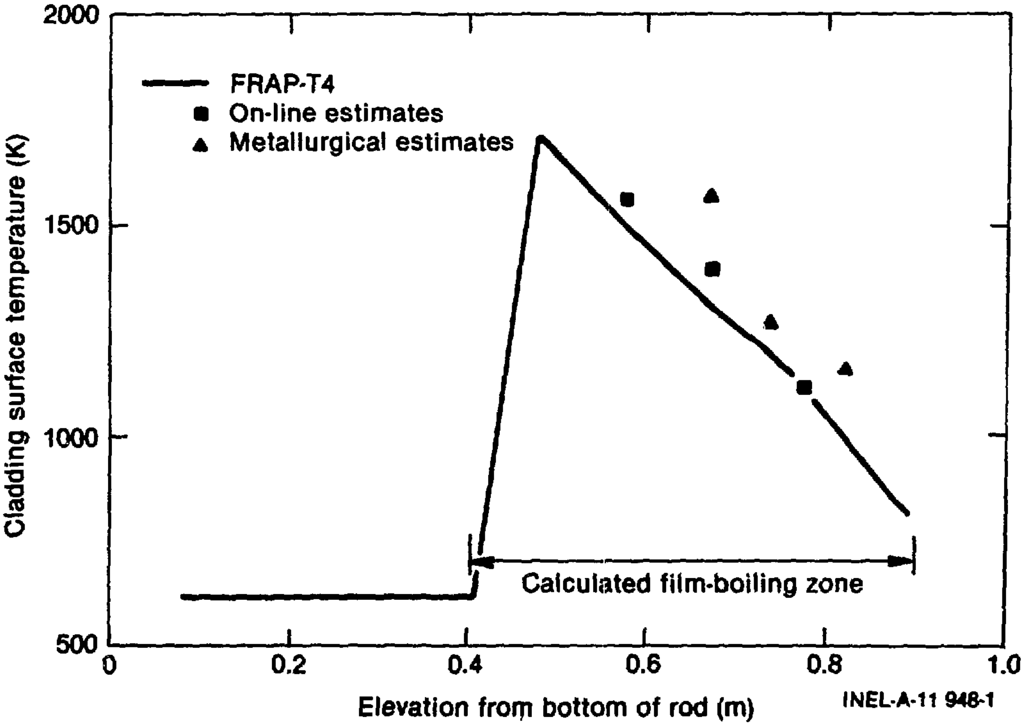

Figure 4. Axial cladding surface temperature profile calculated by FRAP-T4 and measured cladding peak temperatures for the initial film boiling period Test PCM-1.

SESSION XVII

ISSUES WITH RESPECT TO IMPROVED SAFETY

Chairmen

D. Dahlgren - Sandia Laboratories

M. Rosen - International Atomic Energy Agency

KWU's PROTECTED DECAY HEAT REMOVAL SYSTEM

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Erlangen, Germany

ABSTRACT

The design of plant protection against external impacts realized in the standard plant of KWU consists of harmonized combination of administrative, plant engineering and structural protection measure, taking into account the safety-related requirements of the nuclear power plant without impairing the operational conditions. The components necessary to ensure safety functions are situated in different buildings of the plant. To ensure the protection necessary for mastering external impacts taking into account the operational demands and the economic point of view simultaneously, KWU has reflected on the system technique of its standard plants, partially constructing new systems, and especially spatially arranging some systems in a new way to reduce the number of buildings and parts of the plant which have to be protected against external events. With the functional independence and autarky of the provided safety-related measures the administrative and structural measures for protection against third party interaction are supported effectively.

INTRODUCTION

The concept for protection against external impact such as earthquake, explosion shock wave, and airplane crash consists of a co-ordinated combination of organizational, plant engineering and civil engineering protective measures. They fulfill the safety technological requirements of the Nuclear Power Station without impairing the operational interests. According to the type of external impact there are different protective measures:

- An earthquake is a large area impact which affects all sections of the plant as well as the environment of the Nuclear Power Station
- An explosion shock wave and its resulting vibrations is a spatial impact which is mainly limited to the Nuclear Power Station.
- An airplane crash and the resulting vibrations, the impact of wreck parts, fuel blaze and smoke gases represent a locally limited impact which is limited to single buildings of the Power Station.
- Impact by third parties is taken care of by special administrative, organizational and civil engineering measures. This comprises especially the control and aggravation of access as well as the functional independence and autarky of the provided safety engineering measures explained below in more detail.

FUNCTIONS TO BE ASSURED

The consequences of an external impact can differ. They need not necessarily disturb the operation of the reactor or concern equipment which is of safety engineering importance. To trigger the safety systems which are necessary in the case of external impact there are no special "destruction signals" necessary; the available set points of the reactor protection system are used.

The following functions have to be assured for the protection of the environment against inadmissible release of radioactivity also in the case of effects of external impacts:

REACTOR SHUTDOWN AND LONG-TERM SUB-CRITICALITY:

External impacts are usually followed by a setpoint signal (e.g. after failure of feedwater supply setpoint signal "steam generator water level too low"). This setpoint signal causes fast shutdown, also when as a result of the external impact the power supply has been interrupted. This is because the control rods drop - by gravity as sole energy source - into the reactor core and shut it down every time the power supply of their holding coils is interrupted. The reactor is then in the "hot-subcritical" condition.

Also during cooling down of the reactor coolant system the volume contraction of the coolant has to be compensated, and a absorber for compensation of the reactivity increase due to the negative temperature coefficient has to be injected.

HEAT REMOVAL:

The design of the reactor building (including containment, annulus and valve compartment) is such that neither by explosion shock wave nor by airplane crash the function of the equipment to be protected in the buildings will be impaired. Particularly the reactor coolant system remains undamaged. The residual heat removal thus works through emergency feedwater feeding on the secondary side of the steam generator and by blowing off steam into the atmosphere as heat sink. In order to govern an accident independently of the availability of external support a 10 hour emergency feedwater supply necessary for the heat removal procedure described above is stored in the emergency feed building. These supplies have to be renewed before this time limit runs out. The measures necessary for shutdown and heat removal are automatically released so that the reactor remains in a hot subcritical condition for at least 10 hours independently of service and supply.

In contrary to airplane crash and explosion shock wave the impact of an earthquake is not limited to the power plant itself, so that in the case of an earthquake external help cannot be expected within 10 hours. The protection system against earthquakes includes therefore - beyond the systems for securing of the atmosphere as heat sink - also the systems for the direct heat removal from the reactor core (residual heat removal trains). This, however, requires in addition the earthquake protection of the switch gear building, emergency diesel building and the service cooling water system for secured plant.

LIMITATION OF RADIOACTIVITY RELEASE:

The major part of radioactive components is arranged within the reactor building (containment and annulus). This reactor building is protected against all types of external impacts.

The structural design of the auxiliary building is such that a partial protection of the components with radiological importance is provided against external impact, taking especially into consideration fuel blazes after an airplane crash; so the effect of possible destructions in the reactor auxiliary building can kept within the limits acceptable for such accidents.

SYSTEM ENGINEERING EQUIPMENT FOR AIRPLANE CRASH AND EXPLOSION SHOCK WAVE

The turbine building with all its components for "normal" operational heat removal (feedwater storage, feedwater system, turbine-generator, condenser, main steam bypass) as well as the power supply through turbine-generator and the start-up grids are not protected against external impacts. Therefore availability of these parts of the system cannot be taken for granted. In the case of external impact therefore failure of the main heat sink (turbine respectively bypass system with condenser) as well as failure of the feedwater pump system must be assumed. In these cases the energy is removed therefore by evaporation of feedwater and steam blow-off into the atmosphere. The measures to assure functioning of this "substitute heat sink" consist of

- avoiding of uncontrolled evaporation;
  this means the minimum steam generator water level is not to fall below the minimum setpoint and the temperature of the reactor coolant is not to drop too much in order to avoid re-criticality of the reactor;
- ensuring the feedwater supply into the steam generators
- assuring the heat transport from the reactor core to the steam generators by natural circulation inside the primary circuit.

The system parts necessary to ensure these functions are accomodated in different sections of the power plant. In order to protect those system parts which guarantee these functions it would be necessary to protect a major part of the power plant by structural measures. Feasibility of such measures is in some cases technically impossible due to special building structures (extensive and wide halls such as turbine building). Apart from the economical point of view it would be also extremely hindering for the plant operation if major sections of the reactor plant would have to be protected, particularly under the aspect of sabotage protection and its necessary control and aggravation of access. In order to ensure the necessary protection against external impacts and taking into consideration at the same time the operational requirements and observing the economical points of view KWU has re-considered the system engineering in their standard plants. The systems concept was partly re-designed, and in particular their spatial arrangement was changed in order to keep the number of buildings and equipment to be protected against external impact as low as possible.

The basic idea here was to avoid impairing of the primary circuit by such measures as structural protection and consideration of all occurring loads as a result of an airplane crash or an explosion shock wave so that loss of coolant accidents in connection with external impacts can be disregarded. Thus, particularly the requirement of emergency cooling systems functioning and their structural protection could be omitted, which meant also less measuring and control equipment, a reduction in electrical power, and independence of the service cooling water supply. It was possible to concentrate all system engineering measures necessary to ensure the above mentioned functions

- shutdown and long-term subcriticality of the reactor
- steam generator feeding
- necessary isolation measures

within the reactor building and the emergency feed building described below, so that only those two buildings have to be protected against such external impacts as airplane crash and explosion shock wave.

As all required measuring and control equipment, the necessary part of the reactor protection system, and the emergency diesel generators for power supply are also located in the emergency feed building it was not necessary to structurally protect

- turbine building
- switchgear building
- emergency diesel building
- circulation water supply.

This concept for the protection against explosion shock wave and airplane crash does not result in any new points of view as to earthquakes, because here - following the present licensing practice - one still has to start from the basic philosophy that all safety-related features - that means also those for the direct residual heat removal from the reactor core via the residual heat removal system - have to remain available for safety measures in the case of earthquakes.

EMERGENCY FEED SYSTEM FOR GOVERNING ACCIDENTS BY AIRPLANE CRASH, EXPLOSION SHOCK WAVE AND SABOTAGE

The emergency feed system consists of 4 independent trains each of which is clearly assigned to one of the 4 steam generators. Because of the requirement of an independent energy supply in every train the emergency feed pump and the generator necessary for securing the energy supply are driven directly by a diesel engine. This diesel generator can be cooled by the demineralized water feeding the steam generator; so an autarky was made possible. The emergency feed trains are located in the emergency feed building in separated rooms so that, if one train is damaged, no other train can be involved. The emergency feed building also contains the respective storage of demineralized water and diesel oil as well as the measuring and control equipment necessary for governing an accident and automatic start of the required counteractions and the corresponding switch-gears. The design of the systems corresponds in its 4-fold redundancy to the design of the other safety systems. This design was selected independently of a possible discussion about the redundancy required for governing external impacts because the systems located in the emergency feed building are not only used for the governing of accidents resulting from external impacts but also for governing accidents which occur in the feedwater or main steam section of the system itself and to secure the heat removal via the steam generators during the high pressure phase of the emergency cooling case following small break LOCA. Because for these systems the 4-fold redundancy had been introduced already earlier this design aspect was adopted.

With the concentration of all measures necessary to govern "safety cases" in the emergency feed building a clear separation of "operational tasks" and "safety tasks" was possible; the systems for operational steam generator feeding inside the turbine building could be completely stripped of their safety-related tasks and their degree of redundancy could be reduced to the scope which is necessary under operational and availability aspects.

LONG-TERM MEASURES

The protection concept for external accidents is based on securing of "autarky" for 10 hours. For this purpose sufficient demineralized water is stored. The diesel storage is sufficient for a minimum of 24 hours. For replenishment of the demineralized water storage several possibilities are provided, e.g. feeding in of not needed supllies from the demineralized water pools of the emergency feed building, or refill possibility through connection of fire hoses and similar local measures. This gives the possibility of long-term residual heat removal via the steam generators.

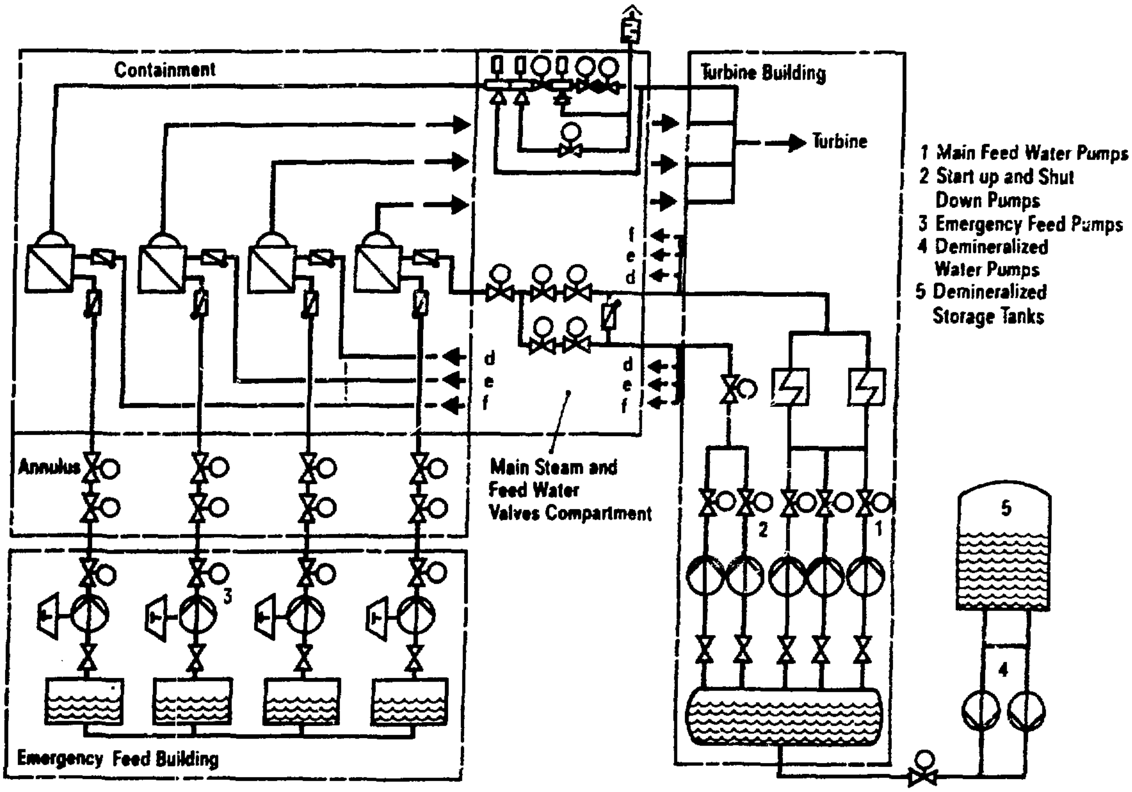
A long-term residual heat removal can, however, also be carried out by means of the emergency residual heat removal chains. Via this 2-fold-redundant system cooling of the reactor during unpressurized conditions and cooling of the spent fuel pool is possible. The changeover from steam blow-off into the atmosphere to emergency residual heat removal with the emergency heat removal chain has, however, the precondition that coolant pressure and coolant temperature in the reactor coolant system have been reduced before, at least to the design values of the emergency and residual heat removal system. This is done manually from the emergency control station.

COOLING OF THE REACTOR DURING UNPRESSURIZED CONDITION OF THE REACTOR SYSTEM AND COOLING OF THE SPENT FUEL POOL

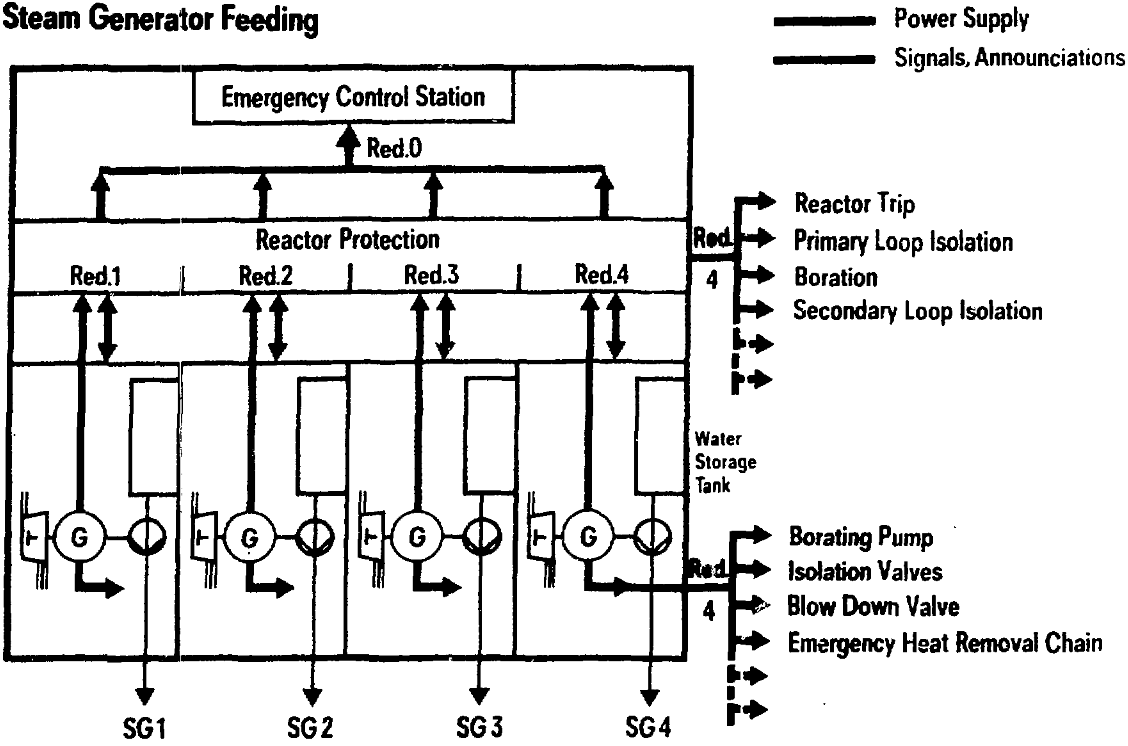
During the refuelling phase the natural circulation in the reactor coolant system and thus the heat transport from the reactor to the steam generators is no more possible. In order to create a cooling possibility for the unpressurized reactor coolant system and the spent fuel pool two emergency residual heat removal chains are fulfilling the following tasks:

- removal of residual heat from the reactor pressure vessel and/or the spent fuel pool in case of failure as a result of external impacts during the refuelling phase,
- cooling of the spent fuel pool in case of a failure during power operation.

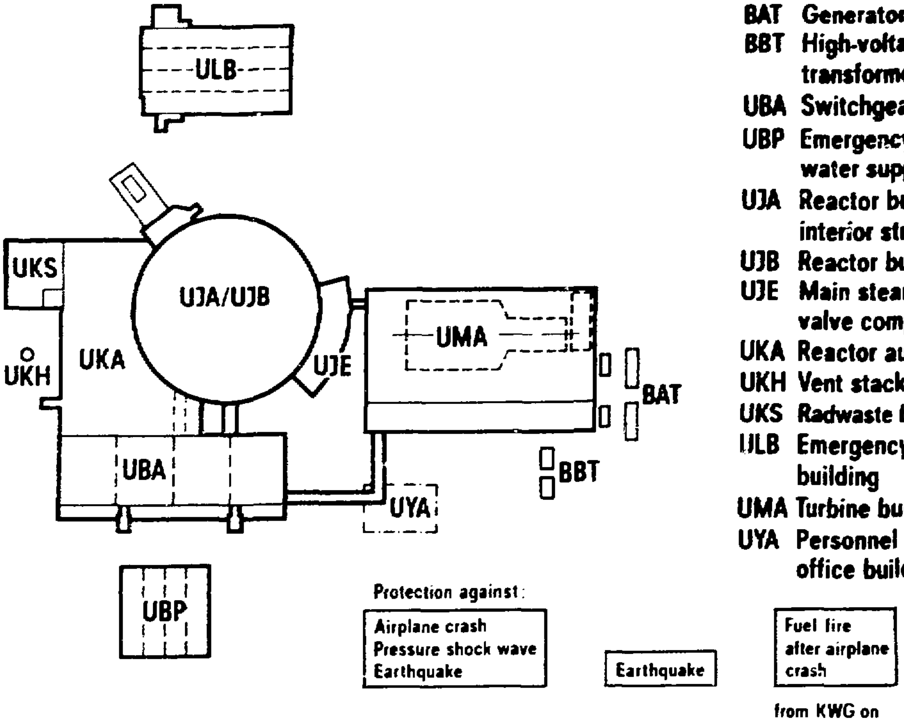
The activation of the residual heat removal chains does not demand automatic measures as there is sufficient time for manual operation after the occurrence of the failure. The emergency heat removal chains differ from the "normal" residual heat removal chains necessary for governing loss of coolant accidents: Only two out of the four cooling chains available are protected against airplane crash and explosion shock wave by according arrangement (sufficient local separation) and according underground routing. In addition the pumps in this residual heat removal chains are adapted to the reduced quantity requirement of the cooling chains respectively the reduced capacity of the fuel pool pumps in order to keep the energy requirement of the emergy feed generators as low as possible.



Main Steam System and Steam Generator Feeding

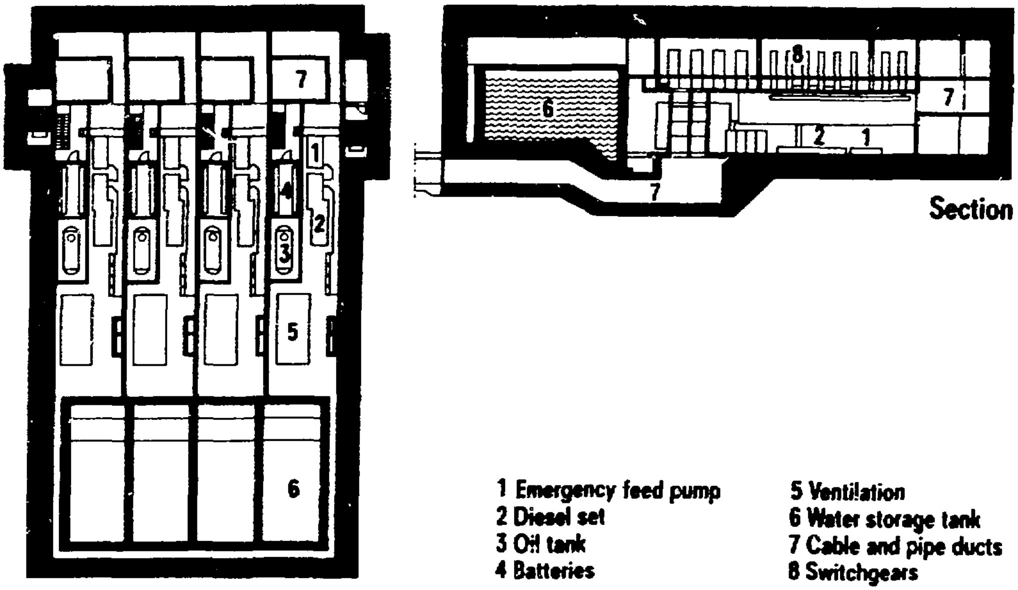


Power Supply and Controls of the Emergency Feed System (schematic)



BAT	**Generator transformers**
BBT	**High-voltage auxiliary supply transformers**
UBA	**Switchgear building**
UBP	**Emergency power and chilled water supply building**
UJA	**Reactor building, containment interior structure**
UJB	**Reactor building annulus**
UJE	**Main steam and feedwater valve compartment**
UKA	**Reactor auxiliary building**
UKH	**Vent stack**
UKS	**Radwaste building**
ULB	**Emergency feed water building**
UMA	**Turbine building**
UYA	**Personnel facilitics and office building**

PWR 1300 MW
Site plan



Section

1 Emergency feed pump	**5 Ventilation**
2 Diesel set	**6 Water storage tank**
3 Oil tank	**7 Cable and pipe ducts**
4 Batteries	**8 Switchgears**

Plan view

PWR
Emergency Feed Building

Dup

SHUTDOWN HEAT REMOVAL SYSTEM RELIABILITY IN THERMAL REACTORS*

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ABSTRACT

An analysis of the failure probability per year of the shutdown heat removal system (SHRS) at hot standby conditions for two thermal reactor designs is presented. The selected reactor designs are the Pressurized Water Reactor and the Nonproliferation Alternative System Assessment Program Heavy Water Reactor. Failures of the SHRS following the initiating transients of loss of offsite power and loss of main feedwater system are evaluated. Common mode failures between components are incorporated in this analysis via the β-factor method and the sensitivity of the system reliability to common mode failures is investigated parametrically.

I. INTRODUCTION

Whenever a nuclear power plant is shut down, there is the need to remove stored and decay heat from the reactor core. Since there is sufficient heat to cause a meltdown of the reactor core even many days after the initial shutdown, it is important to ensure the reliability of the heat removal capability after reactor shutdown.

This paper contains an analysis for the shutdown heat removal systems (SHRS) for selected thermal reactor designs. Particular attention is given to the Nonproliferation Alternative System Analysis Program (NASAP) heavy-water reactor (HWR) [1]. Section II provides a brief description of the SHRS of the NASAP HWR. A fault tree analysis of the HWR SHRS is contained in Section III. Section IV contains a comparative study of the failure rate of the SHRS at hot standby conditions due to a loss of offsite power (LOSP) and due to a loss of main feedwater in the NASAP HWR and in certain pressurized water reactor (PWR) designs.

II. DESCRIPTION OF NASAP HWR SHUTDOWN HEAT REMOVAL SYSTEM

In the first thirty minutes after reactor shutdown while the temperature, pressure, and decay heat are high, the main heat transport system (MHTS) is used to remove residual heat from the reactor. At thirty minutes after shutdown, the reactor coolant system reaches 350°F and 400 psia and the shutdown cooling system (SCS) is used to cool the reactor to

135°F and atmosphere pressure at 4.5 hours after shutdown. In addition to the MHTS and the SCS, the moderator system may be regarded as a large, passive sink for shutdown heat removal. However, in the subsequent analysis, the reliability of the shutdown heat removal system is based on the reliability of the MHTS and the SCS and no credit is given for shutdown heat removal through the moderator system in either an active or passive mode of operation.

II.1 The Main Heat Transport System (MHTS)

The reactor coolant system (RCS) is comprised of two heat transport loops. Each loop contains 370 reactor pressure tubes, two RCS pumps, two steam generators, two inlet headers, two outlet headers, and interconnecting piping and valving. The two loops are connected to a common pressurizer and purification circuit; however, these loops can be isolated if an emergency condition should occur.

The main steam (MS) system transports steam from steam generators to the high pressure (HP) section of the turbine generator. Steam from four steam generators flows through four pipes to a main steam header. Each MS line includes a flow restrictor, power-operated atmospheric relief valve, safety valves, and main steam isolation valve (MSIV).

The main steam system is capable of removing heat from the reactor coolant system following sudden load rejection by automatically bypassing the main steam to the condenser through the turbine bypass system or by venting to the atmosphere through the main steam safety valves or main steam atmospheric dump valves, if the turbine bypass system is not available.

The condensate and main feedwater system returns condensed steam from the condenser while maintaining the water inventories throughout the system. Condensate pumped from the condenser hotwell will then pass through the high pressure feedwater heater to the steam generators.

II.2 Emergency Feedwater System (EFWS)

Upon loss of main feedwater flow, heat can be removed from the steam generator via the safety and relief valves provided that the coolant inventory is maintained by water makeup from the emergency feedwater system. The emergency feedwater system is designed to operate until the reactor coolant system pressure is reduced to a value below which the shutdown cooling system can be operated. The emergency feedwater system pumps unheated water from the condensate storage tank to the steam generators, and is comprised of one motor-driven pump, one turbine-driven pump and a system of piping, valves, and orifices.

II.3 Shutdown Cooling System (SCS)

This system cools the RCS from 350°F and 400 psia at 30 minutes after shutdown to 135°F and atmospheric pressure at 4.5 hours after shutdown. During shutdown cooling, a portion of the reactor coolant will flow out the shutdown cooling nozzles located on the reactor outlet headers. Coolant will be circulated through the shutdown coolant heat exchangers by the low pressure safety injection pumps and returned to the RCS inlet headers.

III. FAULT TREE CONSTRUCTION

A fault tree for the shutdown heat removal system was constructed and is given in Figure 1. In this construction, it is assumed that, at hot standby conditions, the main heat transport system is used for heat removal. At cold or hot shutdown conditions, both the SCS and the main heat transport system can be used for heat removal. The event that a large break at the inlet header prevents the ECCS water from entering the reactor core has also been included in the fault tree. The possibility of loss-of-core cooling capability due to a large number of simultaneous pressure tube failures is also considered. The heat dissipation through the power-operated atmospheric relief valves is included. The indicated transfers 1-4,9,c have been developed further but space does not permit their display. These fault trees can be obtained from the authors upon request.

IV. COMPARISON OF THE FAILURE RATES OF THE SHUTDOWN HEAT REMOVAL SYSTEMS FOR THE PRESSURIZED HEAVY WATER REACTOR AND THE PRESSURIZED LIGHT WATER REACTOR

A comparison is made of the shutdown heat removal system reliability at hot standby conditions of the NASAP HWR and the pressurized light water reactor (PWR). This comparison is made in order to place the NASAP HWR design in the perspective of the more familiar PWR. The designs are similar in the following aspects: 1) both have a high pressure, single-phase primary cooling system; 2) the HWR EFWS is the counterpart of the auxiliary feedwater system (AFWS) in the PWR; 3) both systems have high-head and low-head emergency core cooling systems; 4) the HWR SCS is the counterpart of the residual heat removal system in the PWR. It is interesting to note that recent studies [4,5] have found that there is great variability in the reliability of AFWS among the PWRs themselves. In this regard, the results of these studies may provide guidance to the optimization of the design of the SHRS of the NASAP HWR.

The reliability of SHRS following the initiating transients of loss of offsite power and loss of main feedwater is calculated. Potential common mode failures are analyzed by the β-factor method [2]. As appropriate, where components are expected to be similar, the same reference data (based on WASH-1400 [3]) are utilized. Table I lists the major differences in the SHRS of the two reactors. In these reactors, the feedwater system is used to maintain the water inventory in the steam generator. An auxiliary feedwater system (AFWS) must be able to supply feedwater following a loss of main feedwater supply. The reliability of AFWS was analyzed separately because this is then used in the analysis of the SHRS reliability.

Because of design differences (see Table I), the failure probability of the AFWS in each reactor is different. As indicated in Table I, the PWR AFWS contains one turbine-driven pump train and two motor-driven pump trains, while the HWR AFWS contains one turbine-driven pump train and one motor-driven pump train. Each pump train includes check valves, motor operated valves, manual valves, and pumps. Either of the HWR pumps have the capacity to supply sufficient feedwater to achieve mission success. In the PWR, however, mission success will depend on the sizing of the two motor-driven pumps. The turbine-driven pump can supply

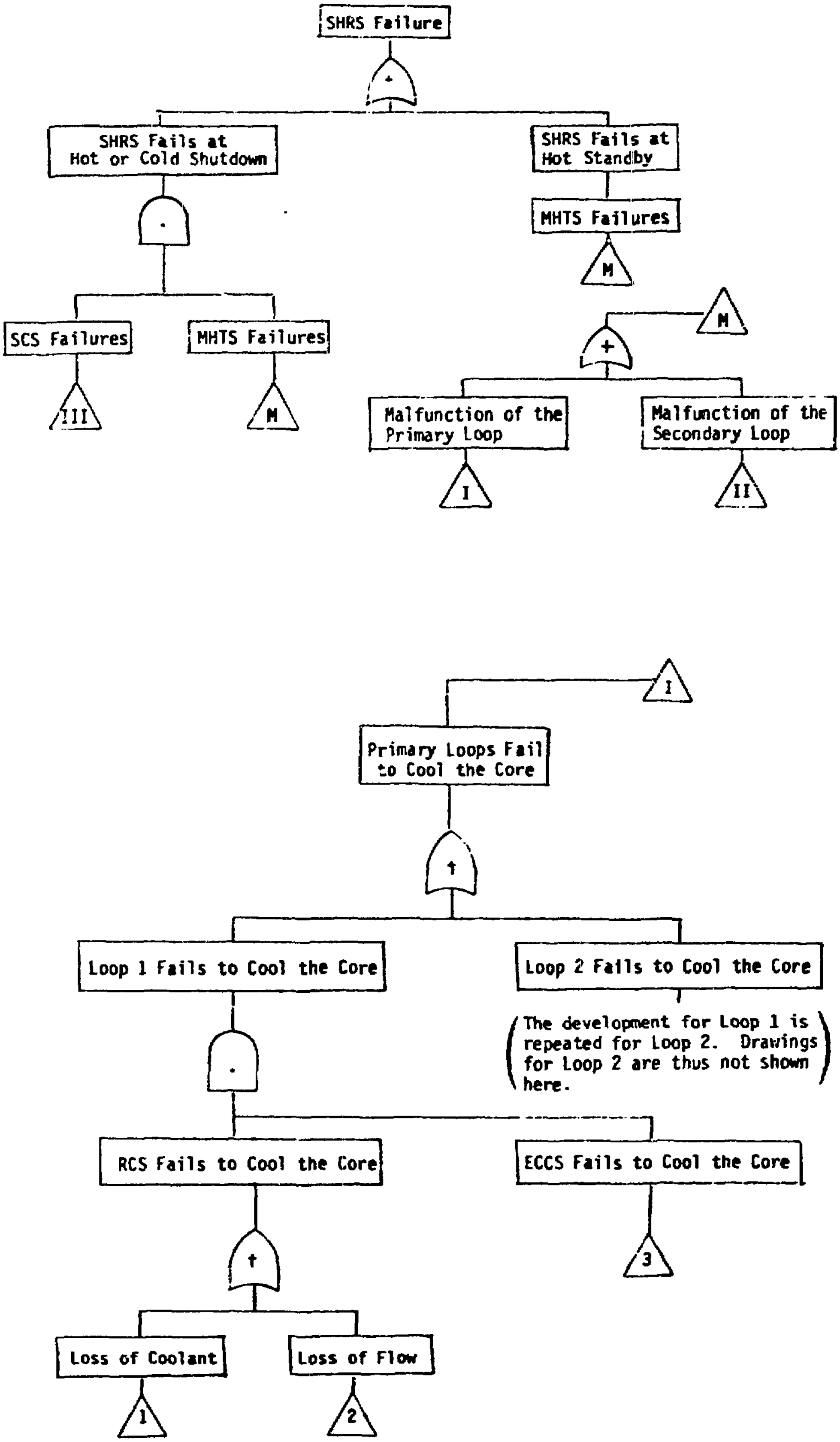
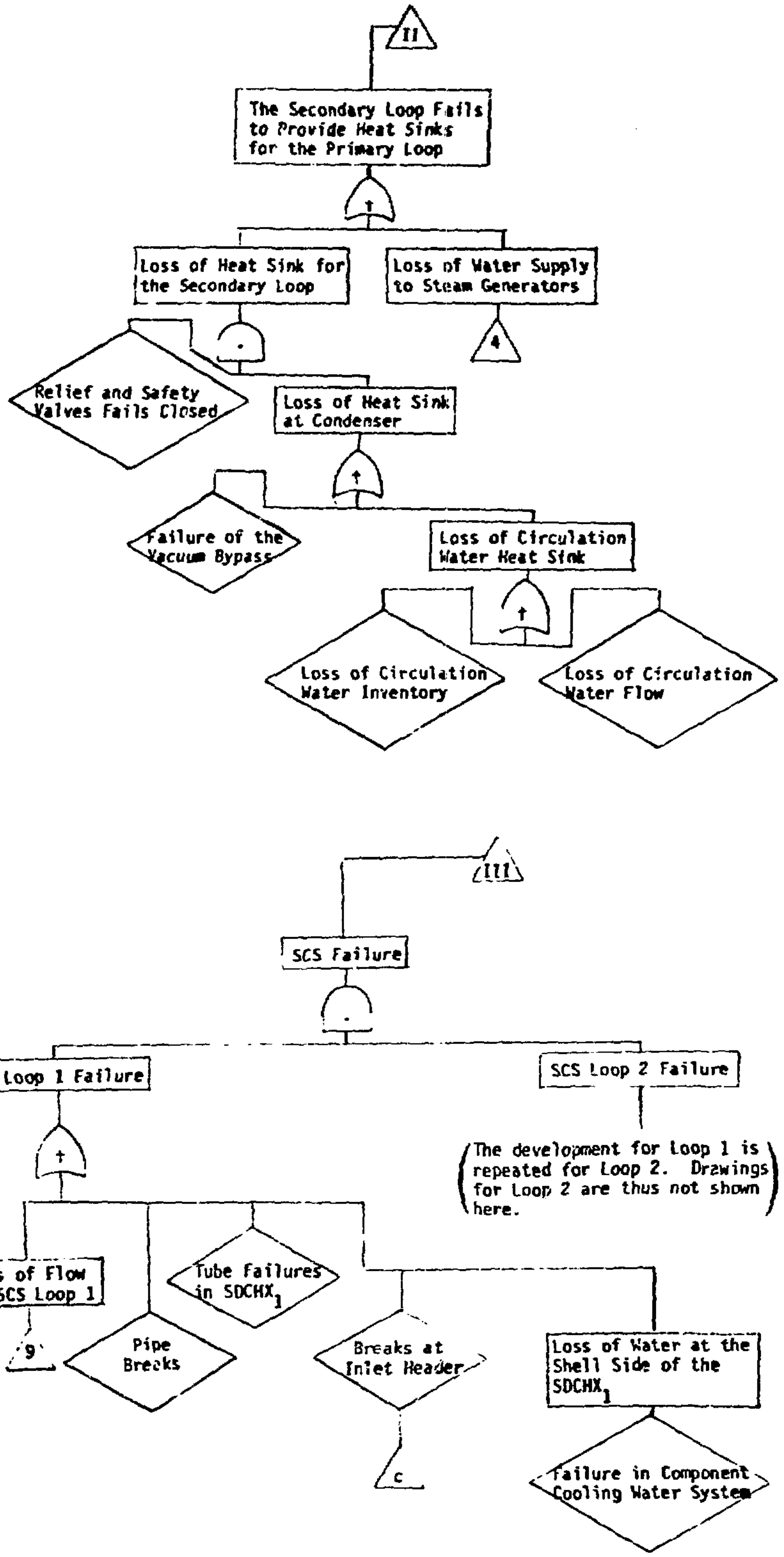


Fig. 1. Fault Tree for the SHRS in NASAP HWR.



SDCHX: Shutdown Cooling Heat Exchanger

Fig. 1 (continued). Fault Tree for the SHRS in NASAP HWR.

TABLE I

Subsystems of SHRS

	PWR	HWR
Active Heat Removal Systems	PHTS MFS AFWS SGS PCS	PHTS MFS EFWS (AFWS) SGS PCS
Passive Heat Sinks	Water in PHTS and SGS	Water in PHTS and SGS Large inventory of water in moderator system.
Other Heat Removal Capabilities	RHRS (for hot and cold shutdown). Assumed natural circulation. ECCS	SCS (for hot and cold shutdown). Assumed natural circulation.* ECCS
Pumps in AFWS	Two motor-driven pumps. One turbine-driven pump. Each motor-driven pump is half-capacity of the turbine-driven pump.	One motor-driven pump. One turbine-driven pump. Both have the same capacity.
AC Power Sources	Offsite power supply. Two diesel generators. One standby diesel generator.	Offsite power supply. Two diesel generators.

PHTS: Primary Heat Transport System
MFS: Main Feedwater System
AFWS: Auxiliary Feedwater System
SGS: Steam Generator System
PCS: Power Conversion System
EFWS: Emergency Feedwater System
RHRS: Residual Heat Removal System
SCS: Shutdown Cooling System
ECCS: Emergency Core Cooling System

*(less certainty because of the horizontal arrangement of pressure tubes and the large contact surface between the coolant and pressure tubes)

sufficient feedwater but either one or both motor-driven pumps may be required to supply adequate feedwater. For example, for the PWR analyzed in Reference 3, mission success was achieved by requiring only one motor-driven pump to supply water. On the other hand, in a recent study [5] it was assumed that mission success required both motor-driven pumps. Therefore, in the present analysis, the reliability calculations will be done both ways: $(PWR)_1$ - requires both motor-driven pumps for mission success; $(PWR)_2$ - requires either motor-driven pump for mission success.

Hardware failures, human error, and unavailability due to test and maintenance of each pump train have been factored into the analysis. The data were adopted from the Reactor Safety Study [3]. The β-factor method [2] was used to relate the possible common-mode events between the manual valves in the motor-driven train and the turbine-driven train and between the diesel generators.

The results for the AFWS failure probability (per demand) with AC power available are as follows:

$$\text{HWR:} \qquad P_H = 1.4\text{x}10^{-4} + (3.0\text{x}10^{-3})\ \beta_v$$

$$(PWR)_1: \qquad P_1 = 2.8\text{x}10^{-4} + (3.0\text{x}10^{-3})\ \beta_v$$

$$(PWR)_2: \qquad P_2 = (3.0\text{x}10^{-3})\ \beta_v$$

Here, β_v is the β-factor for the manual valves and the above expressions are valid for $\beta_v \geq 0.1$.

From these expressions for the failure probabilities P_H, P_1, and P_2 of the AFWS, the failure rate (per year) of the SHRS can be obtained by multiplying P_H, P_1, or P_2 by the frequency of loss of main feedwater. The NASAP study was not performed for a particular site, but it is generally regarded that between one and three loss-of-main-feedwater events per year will occur at a given plant.

If offsite power is not available, then the failure probability (per demand) of the AFWS in each case is as follows:

$$\text{HWR:} \qquad P_H' = P_H + (4\text{x}10^{-4})\ \beta_D + 10^{-5}\ (1 - \beta_D)^2$$

$$(PWR)_1: \qquad P_1' = P_1 + (4\text{x}10^{-4})\ \beta_D$$

$$(PWR)_2: \qquad P_2' = P_2 + (4\text{x}10^{-4})\ \beta_D$$

Here, β_D is the β-factor which accounts for common mode failures between the diesel generators. Note that for the PWRs a third standby diesel is included in the analysis and only the common mode contribution to the diesel generator failure probability is included (for $\beta_D \geq .01$). For the HWR configuration it is seen that common mode failures dominate the diesel failure probability for $\beta_D \geq .05$. The results of these calculations show that the failure of the manual control valves due to a common mode failure (due mainly to human error) contributes significantly to the AFWS failure probability when β_v is greater than 0.1 in both PWRs and in the HWR. Failures of the diesel generators, together with the turbine-driven pump train, do not significantly contribute to the AFWS failure probability unless $\beta_D \geq 7.5\beta_v$. However, for $\beta_v \geq 0.1$, this would imply an unacceptable high unavailability for onsite AC power.

The expressions P_H', P_1', and P_2' can be used to derive the failure rate of the SHRS due to loss of offsite power by multiplying each by the frequency of loss-of-offsite power. Again, generally recognized values are in the range 0.1-0.3 per plant-year.

V. SUMMARY

An analysis of the shutdown heat removal system reliability of the NASAP HWR has been presented and the results have been compared to two variations of the shutdown heat removal systems of the pressurized light water reactor. The β-factor method for quantifying common mode failure has been utilized to compute the SHRS failure rates which result from loss of main feedwater and from loss of offsite power. It was shown that the failure probability of the AFWS in the HWR is bracketed by that of the two variations of the PWR analyzed in this study.

ACKNOWLEDGMENTS

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SAFETY OPTIONS FOR 1300 MWe POWER PLANTS

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ABSTRACT

That safety review of technical options has been performed from end 1975 to mid 1978 through the usual licensing procedure.

The examination of preliminary technical proposals has been resulting in the authorization of creation for the two first 4-loop units and subsequently in the definition and stabilization of main technical options for the ten first 4-loop plants to be installed.

The quantification of probabilistic objectives (first step) led to add news situations inside accident analysis (second step), to define external events to include into the standard design (third step) and to perform a number of additionnal studies (fourth step).

1 - DEFINITION, SCOPE AND MAIN STEPS

The standardization of french power plants requires that prior to regulatory authorizations of creation or first loading the main technical options be reviewed, in order to be detailed and defined for a given type of reactor. This kind of procedure allows to define in advance the features for which detailed studies are necessary to confirm the options which are to be retained.

Within this scope, the review of the technical options of 1300 MWe power plants has been performed from the end of 1975 up to mid 1978, following a path similar to the usual licensing procedure :
- IPSN evaluation,
- review by the groupe permanent,
- notification of requirements to EdF, by safety authorities (SCSIN).

The name "exercice on safety options for 1300 MWe" has been given to the three stages defined hereafter, which have led finally to fix the overall characteristics of the first 8 to 10 units being designed.

a) <u>First step (December 1976 - July 1977)</u>

Prior to the regulatory review of the safety report of the first two PWR 1300 MWe units, the main safety options were defined.

At the end of this step, two letters were sent to EdF by the head of SCSIN to define the elements concerning the probabilistic safety approach to be used, as well as one letter requesting additional informations. Those informations has been subsequently reviewed with the framework of construction permit.

b) Second step (mid 1976 - beginning of 1978)

Authorization of creation of the two first 1300 MWe units (Paluel 1 and 2).

The ordinance of authorization for the Paluel 1 and 2 units has marked the end of this step. As usual, the authorization has been delivered along with recommendations covering the main topics. These recommendations included, among other features, the requests made in the letters of the Head of SCSIN during the first stage.

c) Third step (beginnning of 1978 - mid 1978)

Definition of the main technical options for units to be built in France up to 1982. This last step was requested by the Minister of Industry, responsible in France for the nuclear power plant program. A letter signed by the Minister of Industry defined the characteristic options to be chosen for the first series of 1300 MWe units. Besides, the letter defined the accidents to be taken into account at the design stage.

In many cases, the work performed during these three steps has evidenced a number of safety issues for which additional studies have been requested to EdF and to the builder in order to bring justifications to the safety of the retained options. The results of these studies have been reviewed from the end of 77 up to now, within the regulatory procedures for the authorization of French nuclear power plants.

2 - MAIN TOPICS BEING REVIEWED

Within the short time allowed for the present paper, it's awhward to detail the whole of the points reviewed in France during the "exercise on safety options for 1300 MWe safety".

We shall limit ourselves to the presentation of the main conclusions given to the following features :

- Methodology.
- Technical options related to the containment, to the main coolant activity, to the fuel elements, to the steam generators.

3 - SAFETY METHODOLOGY

3-1.Probabilistic approach and use of quantified goals in safety analysis

The French safety authorities positions are the following :

- Firstly, it is not presently considered that the PWR safety can be demonstrated by probabilistic methods. However, the use of probability should allow to better justify and even to improve the definition and the classification of the "situations" taken into account at the design stage. Probabilistic methods could also be used to improve the definition of deterministic design criteria (for instance, the single failure criterion).

- Two values have been defined by the safety authorities :
  a) The global objective to be considered is about 10^{-6}/reactor-year.
  b) Should a probabilistic approach be used to eliminate an accident condition leading to unacceptable consequences, the corresponding probability shall be proved lower than 10^{-7}/reactor-year.

We shall see later what have been, the conclusions of such positions, up to now.

Systematic studies of the reliability of systems important to safety have been requested from EdF. Actually these ongoing studies are adapted from studies already made for the Fessenheim nuclear power plant (3-loop 900 MWe PWR).

3-2. Definition of accidents to be considered

In addition to the accidents examined up to now (see ANSI N.18.2 list), it was requested that the following accidents had to be studied (disposals should be taken to reduce their probability if so needed or to power their consequences down to a level consistent with the probability) :
- cold overpressure of the primary cooling system,
- failure of auxiliary feedwater system,
- ATWS,
- total loss of ac power supply,
- loss of the ultimate heat sink or of the systems assuming heat transfer towards it.

It has been stated that best estimate assumptions and calculation methods could be used if necessary.

Up to now, the studies related to ATWS, to the total loss of power supplies and to the total loss of the ultimate heat sink have been provided by EdF, as previously stated [1].

3-3. Single failure criterion

During the third step of the exercise, the Institute of Radioprotection and Nuclear Safety has been led to propose a new definition for the single failure criterion. After discussions with EdF and the constructor, and review by Groupe Permanent as the authorization procedure for Paluel power plant, was taking place this definition has been transmitted to the SCSIN at the beginning of 1980. Such a definition can be characterized by the following features :

- for systems permanently or often on request, the application of the single failure criterion is not a sufficient condition,
- the systems important to safety must be designed so that a limited leak (200 liters/minute) occurring at any location could not have unacceptable consequences 24 hours after the beginning of the incident.

3-4. External events

3-4.1. Aircraft crashes

The probabilistic approach, with the threshold previously defined has been retained. It was decided that the value of 10^{-7}/year could be applied to tourism aircraft (airplane mass less than 6 tun), commercial and military aircraft.

In the practice, in view of the characteristics of French sites, only the tourisme aircraft has to be taken into account.

3-4.2. Turbine missiles

The Groupe Permanent has deemed that it was impossible to consider that the probability for the generation of a turbine missile was much less than 10^{-4} per year. To allow for this position, EdF has retained a fan-like distribution for the greatest part of the sites. Where such a distribution is possible.

3-4.3. Explosion hazards

The standard design should withstand an overpressure wave of 50 mbars lasting 500 ms.

3-4.4. Earthquakes

The calculation of the standard design uses the resonator response spectrum defined by NRC in the Regulatory Guide 1.60, according to a maximum ground acceleration of 0.15 g. The average dynamic young modulus is assumed to vary between 5,000 and 100,000 bars.

The sites presently chosen for the construction of the 1300 MWe units have seismics characteristics well within this range. Supporting the unit on antiseismic devices will be provided for sites where the seismic intensity can be higher.

4 - SPECIFIC TECHNICAL OPTIONS

4-1. Containment

The technical solution retained for the containment is a double containment : an internal prestressed concrete wall with no steel liner and an external reinforced concrete wall : thus a total recovery of leakages can be obtained. Studies on this system began in France in 1972 and have shown that the safety characteristics were higher than the single containment with a steel liner with regard to design basis accidents.

In addition to the evaluation of the containment behaviour in the event of a DBA, studies have been undertaken in order to define how would this containment behave in temperature and pressure conditions higher than those generated by a LOCA. Those studies should be completed in 1981.

4-2. Primary coolant activity and protection against radiation

During the first stage of the "safety options exercise", safety authorities have reviewed in detail the problems of worker exposure to radiation due to the primary coolant activity. Additional information has been requested to EdF on the following points :

- possible limitation of corrosion products,
- limitation of the personnel exposure to radiation by detailed planning of the interventions on the main components.

Besides, the safety authorities deem that it is necessary for limiting the radiation exposure that the primary coolant activity due to fission products be strictly limited to a value much less than the value retained for effluent release calculations. This threshold value has not yet been definitely fixed, but safety authorities require to be notified by the utility as soon as the equivalent fraction of rupted claddings reaches 0.03 %.

As concerns the unit design, a strict division of the premises according to contamination hazards has been adopted by EdF.

4-3. Fuel elements

The 1300 MWe units use the Westinghouse 17x17 XLR fuel elements. In order to better assess possible problems of this type of fuel, it was decided that a file related on design characteristics R and D work should be provided every 6 months by EdF and the vendor (Framatome).

The qualification program has been prepared jointly by the utility, the vendor and the safety authorities ; it includes in particular the follow-up of the in pile behaviour of 12 feet 17x17 fuel, which is presently in use for the loading of 900 MWe reactors, and the results of R&D irradiations undertaken in France.

With this strict following-up, any fuel problem should be detected and solutions given early enough. Moreover, fuel operating ranges could be if necessary safety increased.

4-4. Steam generators

Independently of the overall design of EM type steam generators retained in France for 1300 MWe units, the generic issues related to tube corrosion and their behaviour under accident conditions have been reviewed.

The main conclusions are the following :

- studies aiming to the use of materials less sensitive to corrosion than Inconel 600 should be carried on. The present studies deal rather with different treatments than with the choice of different alloys,
- safety authorities have considered that the rupture of SG tubes under accident conditions was not to be excluded and consequently, it was necessary to study the consequences of such ruptures. At the end of 1979, EdF gave the requested information presently under evaluation.

4-5. Rupture in the vessel bottom

Consequences on core cooling of the rupture of instrumentation tubes located under the pressure vessel have been reviewed. EdF provided a file showing that in the present ECCS design the rupture of about 5 up to 10 tubes does not result in unacceptable consequences.

4-6. General implantation

The main characteristics of the 1300 MWe units are the following :

- total separation of the units,
- concentration of engineered safety systems in a specific building,

During the reviewing, the safety authorities deemed that, with these options, the problems of the protection of the components important to safety against malevolant action could be better taken into account (improvement of controls) moreover, a complete separation of redundant channels could be achieved.

5 - CONCLUSIONS

Presently, the quasi-totality of the additional information requested to EdF and to the vendor for the review of safety options has been obtained. Lessons can already be drawn from this work which was made easier in France since there is one utility and one vendor.

- Due to the exact definitions of the field of application of the exercise, an easy under-standing could be found on safety policies.

- Preliminary to any license application, the exercise required that the main options, even not strictly defined, be fully explained to the safety authorities. In many cases, the basis for the choice of the options had to be detailed and then the safety requirement could easily be deduced with full knowledge of the facts.

- Lastly, the whole of the exercise has made it possible to adapt the regulatory reviews, which are basicaly made on a case by case basis, to the standardization of the French units, while not delaying the program schedule.

As a whole, we cannot consider that the exercise has made it possible to solve completely all safety issues, but it helped greatly in promoting the necessary actions.

In the final stage of this work, the TMI 2 accident evaluation had for us the following consequences :

- first some of our positions were strengthened (for instance, on the loss of safety functions),
- and mainly the safety issues reviewed were re-oriented towards the aspects related to the operation procedures, and this had not been considered within the safety options exercise which pertained to the design improvement.

ACKNOWLEDGEMENTS

This paper is based on work performed collectively by all those involved in PWR safety analysis within the Département de Sûreté Nucléaire of the Institut de Protection et de Sûreté Nucléaire.

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ABOUT THE COMPLETE LOSS OF FUNCTIONS ASSUMED BY REDUNDANT SYSTEMS

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ABSTRACT

Are to be taken into account situations resulting from loss of redundant safety systems ?

Two ways of approach were to be probed : evaluation of the failure probability and analysis of the consequences of those situations. The first way leads to improve reliability of concerned systems, the second way to set up mitigating means.

Before TMI-2 occured, safety advices had already been issued about three kinds of situations : anticipated transients without scram, loss of ultimate heat sink, simultaneous loss of out-and inside power supplies.

That, in some cases, something had to be done to improve safety showed the rightness of the concern.

Next step is the study of the loss of both normal and emergency feedwater : The regulatory request has been issued on September 1979.

INTRODUCTION :

WHY TO CONSIDER THE LOSS OF REDUNDANT SYSTEMS ?

Defence in depth, determination of status parameters are absolutely necessary tools in order to identify and analyse events liable to jeopardize the plant.

Nevertheless, such tools turn out to be insufficient to solve two questions :

- to establish the completeness of a list of situations to take into account in order to achieve a given safety degree.
- and mainly to warrant the application of the fundametal safety principle world wide admitted by each nuclear safety body : that is to say that for a given situation, the more serious the consequences are, the lower the probability should be.

It is in such a mind that french safety authorities asked for studies about some situations like the loss of functions assumed by redundant systems, in order to verify that :

- either their consequences were acceptable
- or their probability was low enough.

The first regulatory request is dated October 1976 and has been followed by fuller requests dated July 1977 and May 1978, (see below).

HOW TO GRAPPLE WITH THE PREVIOUS STUDIES ?

Once the previous target had been clearly settled, three main points had to be cleared :

- What criterion has to be retained : conservative or realistic approach ?
- What has to be done if the previous checks turn out to be negative ?
- What functions had to form the subject of such studies ?

As concerns the first point, agreement quickly occured to start the studies using realistic methods and assumptions.

Once criteria have been set up, the second question is quite easy to solve : If the studies turn out to be negative, that means either that consequences are unacceptable, or that the probability is too high : Consequently, improvements have to be implemented in order to reduce the consequences of the situation or its probability : the case has occured (see below).

With regard to the fonctions, the loss of which had to be analyzed, the choice has been turned towards three of them : scram, cooling and power supplies.

The first choice set the wellknown problem of ATWS (anticipated transients without scram). An important potential of studies did exist, constituted by the reports elaborated in several countries, notingly in USA (Nureg 460).

The second choice pointed out the question of the total loss of the ultimate heat sink : the question of the total loss of feedwater (normal and auxiliary) has furtherly been scheduled.

The last choice aimed at a better knowledge of the question of the total loss of in- and outside electrical power supplies. According to what has been done, it can be said that, for that problem, France is ahead of the other nuclear countries.

Obviously, the design situations list could not be considered as completed as far as previous studies had not led to sufficient results.

THE PERFORMED STUDIES AND THEIR CONSEQUENCES

Let us remember that two ways of approach were defined for such studies :
a) to estimate the failure probability, and, eventually, to improve the reliability of thoses systems (way a).
b) To analyse the consequences in order to determine the possibilities of means that could be set up on plants for parrying such failures (way b).
Each studied fonction has to be checked by itself.

Scram Failure During a Transient Needing Scram (ATWS)

The question is wellknown in the States and the french safety advice partly relies upon the content of the NUREG 460.
Up to now, the problem has been treated only for 3-loop plants because the design of the protection system of 4-loop plants was still underway.

The studies have been performed merely following the previous "way a" because of the difficulty to assess the probability. In fact two separated types of analysis have been achieved :

- evaluation of ATWS consequences to determine the most severe transients towards core and primary circuit ;
- reliability assessment of the protection system in order to determine the predominant common mode failures.

First type analysis allowed to work out that "Loss of Feedwater Without Scram" was the most severe transient as regard to primary overpressures and that "Loss of Outside Power Supplies Without Scram" was the predominant transient towards core integrity. Addingly, studies showed a less important pressure pike than Westinghouse computed (cf WCAP 8330), due to a higher capacity of the pressurizer relief valves. On the other hand, the minimum of DNBR is lower, as a result of a lower primary operating pressure.
Studies also showed that there exists a margin towards overpressurization admitted bounds. So is granted mechanical integrity of the primary circuit components.
Finally, the conclusion may be drawn that ATWS do not constitute a fundamental safety problem, insofar as emergency feedwater and, in a less important way, trip turbine are operational.

Second type studies allowed to conclude that there were two main common mode failures :

- If failure occurs in both scram breakers, there is neither scram nor turbine trip : primary mechanical components integrity is however granted, despite of a high pressure pike (220 bars, 3200 psi).
- A common mode failure in relay circuitry cannot be excluded, although logic circuits are slightly contributing to scram failure : So, to diverse actuating signals of emergency feedwater would solve the question.

Safety impact of the previous studies :
French safety authorities required diversification and separation of actuation signals and of their logic treatment to prevent loss of feedwater

without scram and, in a minor way, turbine trip without scram (February 1979).

The operating company, Electricité de France, has already started a practicability study of such improvements allowing to get rid of the consequences of the previous scram common mode failures : such a study is still underway. As concerns 4-loop plants, the previous safety requirements will be already integrated in the protection system design.

Loss of The Ultimate Heat Sink

The studies have been performed following the above "way b" and beared on the assessment of the consequences :

- either of the loss of one of the systems which are necessary to transfer residual heat to the heat sink,
- or of the loss of the heat sink itself without any mind of the origine of the failure.

The encompassing study of the total loss of the heat sink postulates the unavailability of all of the redundant offtakes of water on a site.

It has been agreed that, in such a case, accidental situations have not to be considered : The initial situations are those which are normally met in operating conditions : Full power, hot standby, intermediate standby, cold shutdown with a closed primary circuit, cold shutdown with an open primary circuit and empty fuel pit or filled up fuel pit.

Two main conclusions can be drawn : firstly, the safer withdrawal position is "intermediate standby" because the major difficulty is set by the resistance to temperature of the first seals of the primary pumps, and the maximum allowable increase of temperature ; secondly, the situations which needs the quickest intervening delay is at cold shutdown, on residual heat removal system without the secure of the fuel pit.

According to industrial water supplies which exist on french sites and to the arrangments that are taken, the loss of ultimate heat sink may be stood for a time long enough to restore a sufficient cooling water flow.

Safety impact of the previous studies :

French safety authorities required the inventory of each site water supplies, the disposal of an operating procedure and the results of the experiments presently intended to verify the adequacy of the assumptions concerning the temperature resistance of the pump seals (December 14-1978 & February 15-1979). The procedure, which main features have been checked by tests, is underway.

Simultaneous Loss of Outside and Inside a.c. Power Supplies

The studies have been performed following the above "way b" and beared on the consequences of the simultaneous failure of the auxiliary systems power supplies. But a probability assessment has also been performed and has shown an interesting conclusion : the probability of such an accident has been assessed to "a few 10^{-5}". This value is due for one half to busbars common mode failure and for the other half to supplies failures (grid and both diesel generators).

The encompassing study assumes the total loss of outside supplies (main and auxiliary grids) and the failure of inside supplies (diesel generators). But no additionnal abnormal condition is postulated. This study has lead to analyze the behavior of a plant according to several initial status (full power, hot standby, intermediate standby or cold shut down).

Two main conclusions can be drawn :

- Whatever the initial situation is from full power to intermediate standby (primary temperature > 140°C), primary pressure and temperature control remains possible for a time which length depends on the resistance of the primary pumps seals.
- If the accident occurs at intermediate standby (primary temperature > 140°C) or cold shut down, the development of the situation depends on the water level above the core.

That means that, above 140°C, the safer withdrawal position is intermediate standby and that the case for which the intervening time is the shortest is when the minimum level above the core (shutdown and SG tubes inspection).

Because the injection to the primary pumps seals is secured by a turbine driven pump fed with secondary steam, french 4-loop plants may stand the situation in a better way than 3-loop plants : Control remains possible for almost 20 hours in the first case (then the pressurizer gets filled up) against around 1 hour in the second case (then pressurizer is empty and primary coolant is boiling).

Safety impact of the previous studies :
French safety authorities stated that 20 hours are enough time to allow to recover the plant control, that is to say to recover at least one power supply, inside ou outside (December 14-1978).

They also stated that 1 hour is too short time and that specific disposals have to be taken to extend the respite to a length of the same range than previously (February 15-1978).

Elsewhere operating procedures are underway in both cases and studies of disposals are already undertaken.

FROM A REGULATORY POINT OF VIEW

The question of the study of the loss of redundant systems has been clearly set by Nuclear Safety Department on October 1976, taking the opportunity of the definition of future 4-loop plants safety options [1]. Groupe Permanent (french safety advisory group) stated the advisement on December 1976.

Service Central de Sûreté des Installations Nucléaires (Nuclear Facilities Safety Central Service) presented the regulatory requirement on July 1977, settling the global level of risk and probability thresholds [1].

The requirement elsewhere specifies that "Simoultaneous loss of major safety redundant systems has to be checked case by case in order to determine wether it had to be taken into account in the design".
The double way of approach (a and b) was clearly set and the first field of study was specified (the three previous cases).

Groupe Permanent stated twice on the results of these studies first time (December 1978) on 4-loop plants studies ; second time (February 1979) on 3-loop plants studies.

From a regulatory point of view, does it mean that loss of redundant systems is to take into account in the design ?

The answer is given by a recent letter, issued from French Ministry of Industry on September 1979, that draws up the general lines following which future 4-loop plants will be designed : at the end of section I-1.B. it can be read that :
"In addition, are underway the studies of the following situations with regard to which, if necessary, specific arrangements will be implemented to lower their probability or to mitigate their consequences at a level in connection with that probability :

. (...)
. Failure of emergency feedwater system under frequent transients needing it.
. Failure of scram system under transients needing scram.
. Complete loss of a.c; power supply.
. Complete failure of the ultimate heat sink or of systems assuming heat transfer towards it".

CONCLUSION

Is the loss of redundant systems to be taken into account ? The question has been set. Studies have been worked up in order to assess probabilities and consequences. Disposals have been considered in order to cope with such failures or to mitigate their consequences.

In fact the actual target was more complex. Besides the previous purpose the objective was dual :

- firstly, to make sure that not to take a certain type of accident into account at the design level was cogent, and to provide eventual easy measures of ultimate help,
- secondly, to improve our knowledge about the behaviour of pressurized power plants under unusual conditions and simultaneously to prepare operators to react with adequacy to similar situations.

The present conclusion that can be drawn from the studies already performed is .that the care of safety authorities to make sure they were not leaving a weak safety area in the background was a right care, insofar as it allowed to point out that in some cases something had to be done to improve safety.

In other respects, it can be stated that those safety improvements do not settle a prohibitive increase in costs especially when they are provided at the design level.

TMI-2 accident constitutes obviously the most important feature, since the end of the "Basic Safety Options" [1] exercise : the analysis of the accident reinforced safety authorities in the conviction they were on the right path and they had to carry on.

So did the regulatory safety authorities.

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"(PROTECTION) LIMITATION SYSTEMS"

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ABSTRACT

Since March 1977 German NPS can be designed under an improved "Defence-in-Depth Concept" because of the inclusion of Limitation Systems in the German Licencing Rules.
History and survey of "Limitations" were given in Brussels 1978.
This paper describes the main features of the:
"Power Density Limitation for the Core Top Half"
with graded quality requirements for graded functions

- protection limitation against center line melting
- condition limitation for maintaining
    - reactor power Initial Conditions (IC's) below Loss of Coolant Accident (LOCA) limits
    - local power changes below fuel design limits
    - DNB-Ratio above Loss of Flow Event (LOFE) limit
- operational limitation for power distribution control assistance

These functions are used as examples to show the contribution of Limitations to the "Overall Operational Safety" as required in the Kemeny-Report.

INTRODUCTION

As a turnkey contractor of Nuclear Power Stations (NuPowS) the German KRAFTWERK UNION AG (KWU) has gained in the past considerable knowledge to accomplish a high degree of

"Overall Operational Safety"

which is required as the main goal in design and operation of NuPowS by the US-President's Commission on "The Accident at the Three Mile Island" (Kemeny Report).

One of the features to approach Overall Operational Safety is the introduction of "Limitation Systems" in the operational area between the (mostly Reactor-) Feedback-Controls and (classic) Reactor Protection System (ReProS) which all together then represent a "Defence-in-Depth Concept".

Limitations combine intelligence features of feedback-control systems with the high reliability of protection systems.

HISTORY

The first Limitations were introduced in the Stade (1972) and Biblis (1974) type plants - at a time when distinct Licencing requirements for them did not exist. The objectives were:

- to limit operational disturbances to levels below the trip level, to avoid unnecessary scrams,
- to prevent operational events from growing to accidents, thereby improving the plants' availability and preserving the plant from avoidable stresses [1].

In the meantime this contribution to the "Defence-in-Depth Concept" was acknowledged by the German regulating body "Kerntechnischer Ausschuß" (KTA), who included the Limitations into the KTA-Rule No. 3501 (for design of ReProS) issued March'77. This Rule in accordance with IAEA-and IEC-Rules recommendations, then led to a further broadening of scope of Limitations and their qualification in the plants under construction.

Details of philosophy, functions and history of limitations (for the scope at the end of '78) were reported during the last ANS/ENS-Topical Meeting at Brussels [2].

THE EXAMPLE OF THIS REPORT

Reactor core protection was from the very beginning one main objective for safety considerations. In small, stable reactor cores designed for constant load operation such systems could be designed very simple and easy. However with large, unstable cores and with NuPowS designed for arbitrary load follow operation new solutions were necessary.

Theoretical and practical experience has shown that only a combination of

- an advanced, fully automized Power - and Power Distribution Control
- a Limitation of Reactor (Integral) Power
- a Limitation of Power Distribution
- a Limitation of Rod Bank Movement together can solve

indeed all the problems of large core protection.

In modern 1300 MWe-KWU-NuPowS about 20 Limitation Modules are installed.- As an example of the development and final layout of such modules the

"Power Density Limitation for the Top Core Half" or

"Top-Peak-Reactor Power Limitation = "Top-Peak-RePoL"

will be described in more detail.

This system limits the peak power density in the top half of the core. It is one of the most advanced Limitation Systems and performs operational and safety-related functions.

CONTROL CONCEPT

To understand the (later described) function of the Top-Peak - RePoL the core cross-section with the control rod banks and the in-core power distribution detectors should be know before (Fig.1)

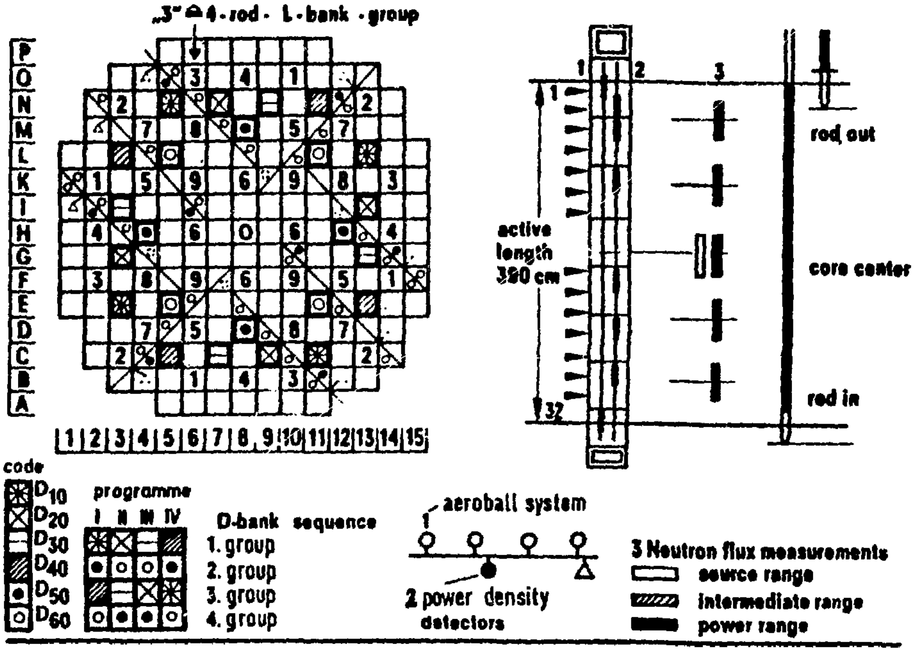

Fig.: 1 Core cross-section of large modern KWU-PWRs

The "Combined, fully automized Reactor Power and Power Distribution Control" for 1300 MW-NuPowS uses the following control elements:

- a strong (ca. 50 rod) bank; only ca. 0 to 10 % inserted
  with strong axial power distribution
  and also (some) coolant temperature control } features

  called "L-Bank"

- a weak (4x4 rods) bank, the four rod groups of which are inserted one after the other if decreasing load slowly from 100 % to 0 % (figure 2)
  with (mainly) coolant temperature control
  and (as a by-product) Doppler feedback compensation } features

  called "D-Bank"

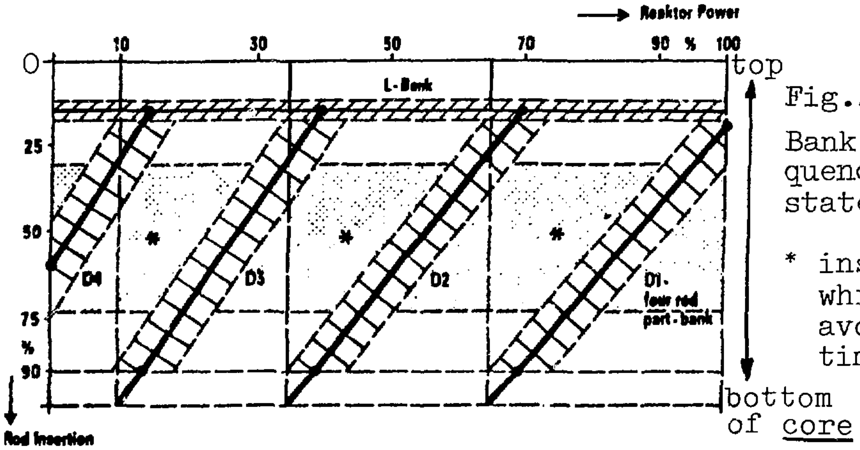

Fig.2:

Bank insertion sequence in steady-state operation

* insertion depths which should be avoided for long times

- boron and demineralized water control valves with very slow coolant temperature control features, also exercising integral Xenon and burn-up compensation.

  (Quick Xenon burn-out and build-up as well as coolant temperature decreases of all velocities are of course used as inherent reactivity sources of the overall automatic control system).

The above mentioned control concept(which cannot be described here) was developed by means of a self-constructed one purpose computer with an analog 80-zone-core-model of varying arrangement [3] together with 20 000-node quasi-stationary physics models.

INCORE DETECTOR SYSTEM

The information on the actual power density distribution is provided by incore detectors of the self-powered type, utilizing cobalt as detector nuclide. These detectors are quick, reliable and powerful enough to give the required information.

In the core cross-section lances with detectors are arranged in 8 radial and azimuthal positions:

2 in the core central- and 6 in the peripheral zone.

In each lance six detectors are non-equidistantly located:

3 in the top- and 3 in the bottom half of the core.

It has been shown in the past that the information of this 48 detectors combined with the capability of the Bank Movement Limitation System, which prevents the rod banks from spurious movement, is sufficient for correct control and limitation functions.

THE-TOP-PEAK-REPOL

The Top-Peak-RePoL is a redundant (4 channel) system which initiates protective actions aimed at bringing back the peak power density into the normal operating range, if operational limits are exceeded.

Measuring Subsystems:

The 24 Power Distribution Detectors (PDD's) of the top core half are all sensors of all four channels of the "Top-Peak-RePoL" (decoupled signal distribution). They are calibrated to the highest power density of their " surveillance zones" (by use of plant process computer).

The highest signal or-at a larger than specified permissible deviation of the highest from the second highest signal-the latter one is used for further processing as the main control variable. Some other variables like integral reactor power, coolant inlet temperature and -pressure and main coolant pump speed (all are used four times redundant) are being used for signal correction and actuation value computation. (Fig. 3)

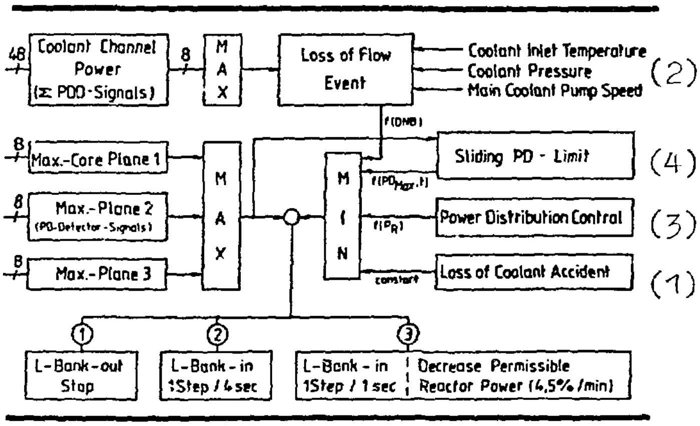

Fig.3:
Survey of the Power Density Limitation for the Top Core Half

Initiation Subsystem:

The above described main control variable is compared to the reference value of the system; in case of overriding protective actions are initiated.
The reference value is generated of four main terms each of them representing a separate operational or safety-related function: (Fig. 3)

1. a) To avoid fuel center-line melting during anticipated operational occurrences;

   b) To keep the energy stored in the fuel pins below the value which is assumed as an initial condition in the LOCA analysis.

   This term is a high, constant value.

2. To ensure acceptable initial conditions with respect to DNB during a Loss of (partial) Flow Event (LOFE). The assigned peak power density limit depends on the actual coolant conditions and on the actual power density for all three planes of upper detector locations.
3. To ensure an acceptable axial power shape during part load operation (bottom peak!) so that fast return to full power (by withdrawal of rods) is possible without new Top-Peak-RePoL action. The corresponding peak power density limit is dependent on integral reactor power.
4. To meet mechanical fuel design limits with respect to local power changes. The corresponding peak power density limit depends on the long time local power history of the highest measured value.

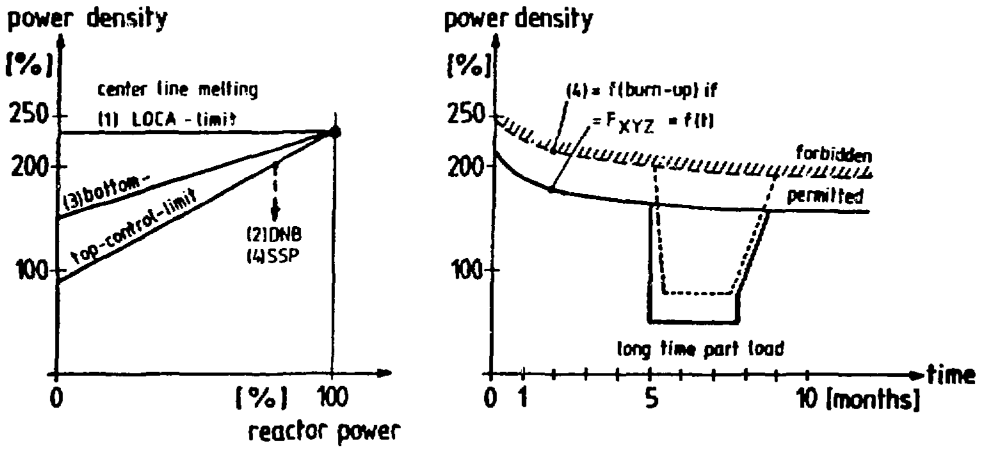

Fig.4: Initiation Channels Reference Values

QUALIFICATION

Function (3) is related to control and therefore to plant availability only and require a normal quality standard.

Function (4) contributes to avoid unnecessary material stresses. In positive understanding of "Operational Safety" it is a (yet non strong required) "Grey" safety-related equipment (Condition Limitation in the German KTA 3501 sense. Category 1 - equipment of the IAEA-Safety Guide D 8).

Functions (2 and 1 b) constitute, what the German Rules really call a "Condition Limitation", the equipment of which has to meet more elaborate quality standards than normal (e. g. redundant design, careful documentation and maintenance, but not worst-case design).

Function (1a)constitutes what in Germany is called Protection Limitations. The quality requirements for these functions are the highest of all mentioned and determine therefore the quality standard of the whole Top-Peak-RePoL equipment. The required quality standard is the same as that of a classic protection sytem initation channel.

Actuation Subsystem:

After the comparison of control variable and reference value three different measures are actuated due to the size of the control deviation:

- Blocking of L-Bank withdrawal so the quickest and strongest source of axial power shape deformation is eliminated
- Insertion of the L-Bank with low speed (ca. 1 step-of 1 cm-in 4 seconds) influencing the power shape to decrease power density in the top core half. This is a strong effect and the intended main action.
- Insertion of the L-Bank with maximum speed (1 step per second) and reduction of the signal "(Maximum) Permissible Reactor Power", the reference value of the "Reactor Power Limitation", with a gradient of 4,5 %/min.

The last two mentioned protective actions cause in most cases intelligent interaction with other safety systems of the plant:

- either with the "Bottom-Peak-RePoL": if after an "L-Bank-insertion"too big a bottom-peak occurs. The Integral Reactor Power must be decreased by D-Bank-insertion, or Rod-(Pairs)-Dropping is initiated (which cannot be described in more detail in this context)
- or with the (Integral) "Reactor Power Limitation" the reference value of which is the "(Maximum) Permissible Reactor Power" signal which is addressed above.

DEACTIVATION

After de-activation of all initation criteria all actuated measures stop to work and the operational feedback control activities which were overridden before by the limitation actions try to bring the plant automatically back to the (more optimized) conditions before the event.

If the prime cause of the event is still valid, the limitation action will stay active. This is annunciated to the operator while the plant is running in a suboptimal mode. If the disturbance was spurious the plant returns automatically back to normal operation.

ADVANTAGES

Using the example of the "Top-Peak-RePoL" some advantages of the use of Limitations can be explained by showing their big contribution to the Overall Operational Safety:

Economical :

- they permit the plant to be operated more flexible and closer to design limits without loosing safety
  - by using incore-detectors with their more detailed information of the core status (for controls and limitations)
  - by allowing application of more optimized and sophisticated (and therefore mostly less reliable) control systems
- they "help to avoid scrams" in a real "defence-in-depth-concept"
  - by multiple staggered use of control, limitation(with fast setback functions)and only at least shutdown actions.

Safety-related (in an "operational safety sense"):

- they decrease component stresses
  - by avoiding scrams and
  - by first using limited set-backs (causing smoother transients)
- they reduce the probability of human errors
  - by early, differenciated counteractions giving time and therefore confidence to the operators/optimizers
- they simplify the licencing procedure
  - by assuring the dedicated initial conditions of the Safety Analysis by their functions as Condition Limitations.
  - by suppression of a lot of otherwise possible sources of disturbances (e.g. reactivity disturbances) being redundant in design and therefore higher failure-resistant.

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BACKUP SAFETY SYSTEM BALL REPLACEMENT PROGRAM PHYSICS TESTING AT THE HANFORD N REACTOR

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ABSTRACT

A series of measurements established the acceptability of B_4C-carbon balls as a suitable replacement material for the Sm_2O_3-alumina balls in the N Reactor backup safety system. Out-of-reactor tests provided a measurement of the required boron loadings of the new balls, confirmed superior burnout characteristics, and verified significantly less material activation. The in-core testing measured the shutdown margins for the backup safety system with the B_4C-carbon balls under normal and special failure conditions. As a result of the ball replacement program, improvements in safe operation and better radiological control should be achieved at the N Reactor.

INTRODUCTION

During the last extended maintenance outage at the Hanford N Reactor, the special neutron absorbing balls were changed out in the backup safety system. Associated with the change out were extensive out-of-reactor and in-reactor testing programs to assure that the replacement balls provided equivalent shutdown characteristics to the original balls.

N Reactor was designed with a dual shutdown safety system: the horizontal control rods and the ball backup safety system. The two systems have independently the capability to shut the reactor down sufficiently fast to prevent any fuel damage during postulated reactivity transient and assure that the shutdown reactor will remain subcritical for all credible accident conditions. The backup safety system was designed to operate with balls made out of sintered Sm_2O_3-alumina. The balls provided the required neutron absorption, heat resistance, and the needed structural integrity. However, potential radiological problems associated with prolonged irradiation of ball material in the reactor were not anticipated. The radiological problems, the unfavorable nuclear burnout characteristics of samarium, and the cost of replacement balls led to the decision to replace the original balls with balls fabricated out of more suitable materials.

N REACTOR AND THE BACKUP SAFETY SYSTEM

The Hanford N Reactor is a 3860 MW(th), graphite-moderated, pressurized-water reactor operated by UNC Nuclear Industries for the U. S. Department of Energy. It is called a dual-purpose reactor since the waste heat developed in producing nuclear products for the Department of Energy is used to generate up to 860 MW electrical power for use throughout the Pacific Northwest. The reactor core is a graphite cuboid approximately 10 m square at the face and 12 m long. A total of 1003 horizontal Zircaloy-2 pressure tubes penetrate the graphite stack. Perpendicular to the pressure tubes are 86 horizontal control rods used for normal control and for shutdown of the reactor. Vertical to the pressure tubes are 107 channels which can be filled with neutron absorbing balls to provide backup shutdown capabilities.

A simplified view of the reactor core showing the major penetrations through the moderator is shown in Figure 1.

The ball safety system is the backup reactor shutdown system. Each of the 107 ball hoppers contain 0.064 m^3 of balls made out of special neutron absorbing materials. The entire ball system is automatically activated if: the rod system fails to insert in a prescribed time period, a seismic trip occurs, or the emergency core cooling system is activated. The balls flow by gravity from the hoppers into the vertical channels penetrating the core. Within 47 seconds, the exit piping and the ball channels are filled. After the control rods have been inserted, the balls can be circulated back to the top of the reactor and filled into the hoppers. A typical reactivity insertion curve for the ball system is shown in Figure 2. The initial curtain of balls will terminate all postulated reactivity transients, and the filled ball channels will provide the required subcritical shutdown margin for all postulated accidents. The original balls were approximately 0.95 cm in diameter, sintered Sm_2O_3-alumina balls. The replacement balls made out of baked B_4C-carbon were designated to be 1.11 cm in diameter to permit mechanical separation of old and new balls. Figure 3 shows a sample of Sm_2O_3-alumina balls and rounded B_4C-carbon balls.

REASONS FOR BALL REPLACEMENT

During the 17 years of N Reactor operation, some distortions in the graphite stack have occurred, which have resulted in the leakage of Sm_2O_3-alumina balls from the ball channels into the reactor core. Any balls exposed to a thermal neutron flux of 10^{13} - 10^{14} n/cm^2-sec become highly activated after a short period of time. The activity following irradiation is due to the decay of Sm-153, Eu-154, Eu-155, and Eu-156, and high dose rates from such materials will persist for years. The burnout of any balls lost in the reactor core will be very rapid initially as the Sm-149 is depleted but then, due to isotopic transmutations, levels out to a very gradual change, as seen in Figure 4. The adverse impact from the activation and slow burnout of the displaced Sm balls precipitated an alternate control material replacement program for the N Reactor backup safety system. A ball material was needed which provided equivalent neutron absorption characteristics to samarium, could be readily burned out by neutron capture if lost to the stack, and would not experience activation of long half-life daughter products emitting high energy gamma rays. The suitable replacement material was identified as

B_4C in a carbon matrix. The required concentration of boron in the replacement balls equivalent to the control strength provided by the samarium balls was calculated and confirmed by neutron transmission measurements using a Cf-252 source.[1] Tumbled B_4C-carbon right circular cylinders were ordered, received, and tested by neutron gauging to demonstrate that they contained the required concentrations of boron. Additional grinding of the balls was necessary to produce more rounded shapes to facilitate the flow of the balls in the handling system. Selected ball samples were neutron gauged for a second time to assure that the grinding operation did not decrease the ball absorption characteristics.

Preliminary testing included irradiation of samples of the B_4C-carbon balls in the reactor under typical flux conditions to determine residual radioactivity. The observed results confirmed that the B_4C-carbon ball activity would be at least three orders of magnitude less than for the Sm_2O_3-alumina balls irradiated under the same conditions. The ball material burnout was studied analytically. The replacement balls would burn out rapidly in the reactor, while the original ball material would initially burn out rapidly and then remain at a nearly constant neutron absorption level for years, as shown in Figure 4.

IN-CORE TESTING PROGRAM

The out-of-reactor testing established that the ground B_4C-carbon balls had adequate neutron absorption characteristics as a replacement for the Sm_2O_3-alumina balls and that the B_4C balls exhibited significantly improved burnout and activation characteristics. An in-core testing program at low power conditions was undertaken to measure a system worth for the B_4C-carbon balls and provide answers to specific safety considerations associated with the ball system performance.

The in-core testing was divided into three parts. This first part measured the localized reactivity and flux perturbation caused by a single channel of balls in the reactor. The second part was a ball system worth measurement obtained by filling every other ball channel until the reactor was just critical with all the control rods withdrawn. As part of this measurement. the reactivity worths of groupings of ball channels in different regions of the core were measured. The third part of the test measured criticality for the largest number of contiguous empty ball channels in the highest flux region of the reactor with no control rods in the core. The test measurement provided important information on the number of ball channels out-of-service in the most adverse configuration for total control administration. All reactivity measurements were accompanied by detailed in-core flux traverses obtained with flux wires and a movable fission chamber. The two critical configurations with no rods in the reactor are shown in Figure 5.

The measured reactivity worth for the alternate ball channel configuration was 31.1 mk, compared to a calculated value of 33 mk and a worth of 29.2 mk assigned to the Sm_2O_3-alumina ball system. In all instances, good agreement between calculated and measured reactivity worths was observed. The in-core measurements confirmed that the use of the new B_4C-carbon balls in the backup safety system provided an improved shutdown margin as compared to the original balls. The total control concept for the ball system was validated, and a calibration of ball channel worth in different regions of the core was obtained.

CONCLUSION

As a result of the out-of-core and in-core testing program, the backup safety system with the B_4C-carbon balls provides improved shutdown control, superior burnout characteristics, and significantly less material activation. The new system will contribute to safe operation and better radiological control at the Hanford N Reactor.

ACKNOWLEDGEMENTS

The contributions of L. R. Dodd, J. H. Eaves, R. L. Moffitt, and other members of the Reactor and Applied Physics Staff are recognized for their contribution to this effort.

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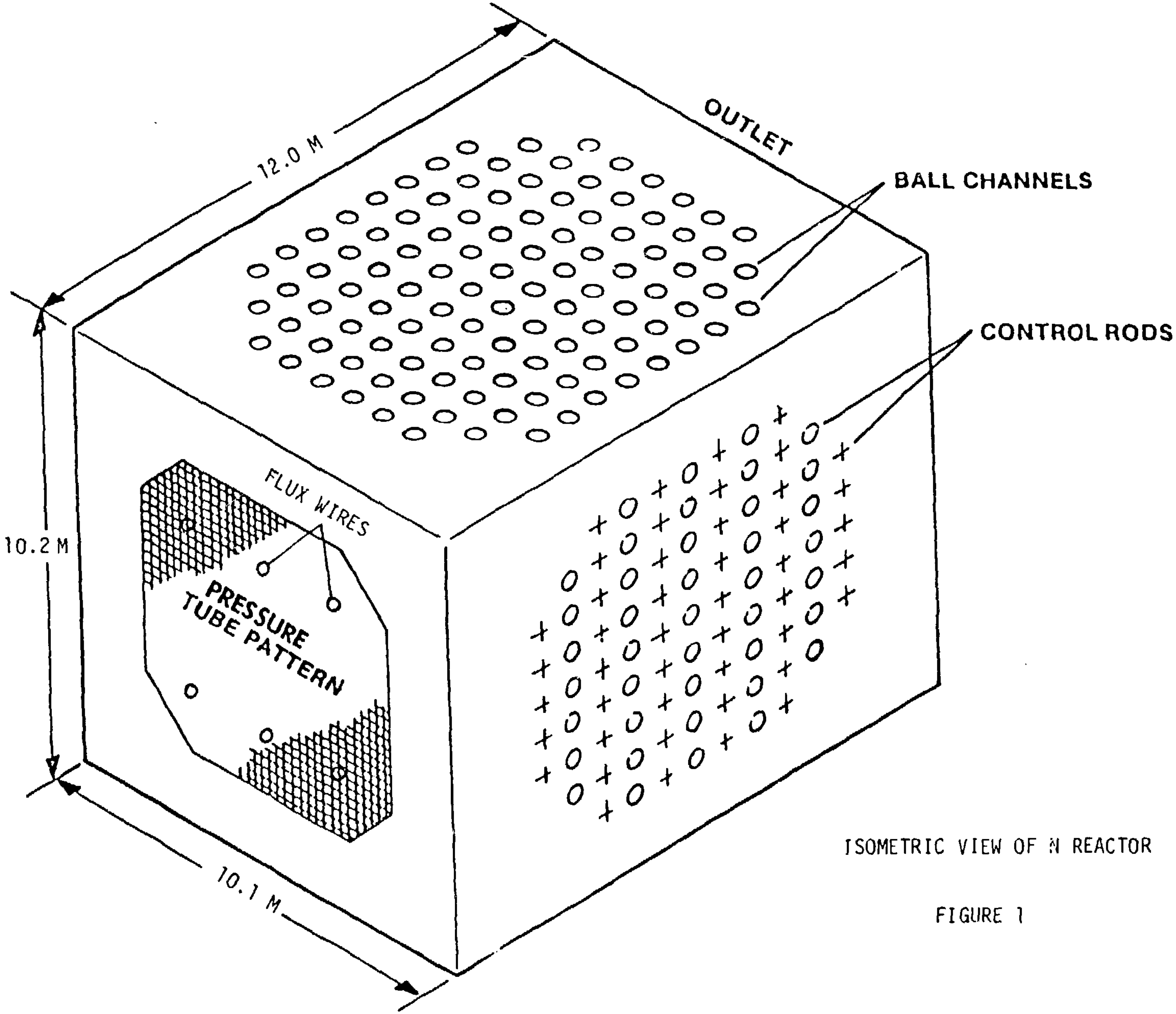


ISOMETRIC VIEW OF N REACTOR

FIGURE 1

Fig. 1. Isometric View of N Reactor.

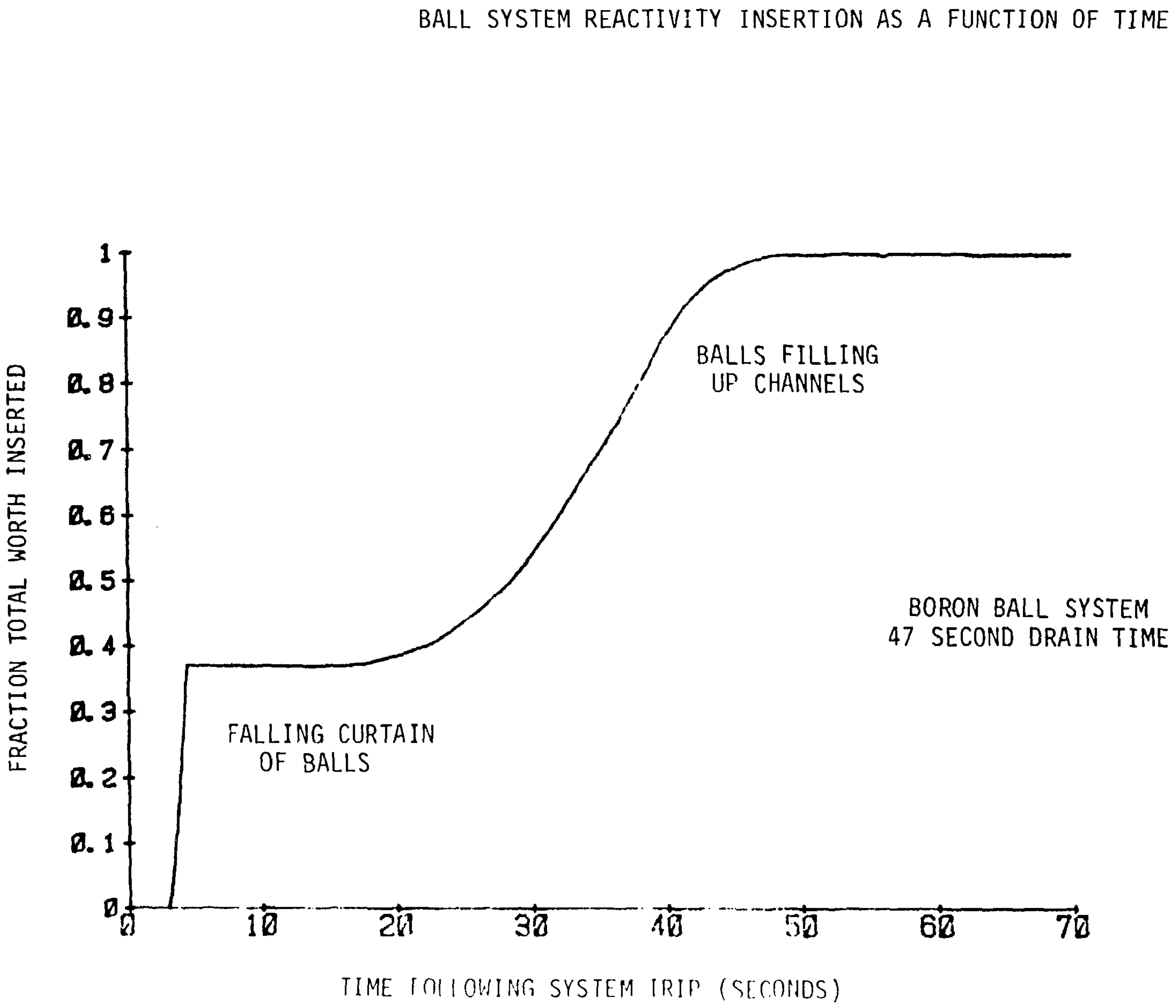


FIGURE 2

Fig. 2. Ball System Reactivity Insertion as a Function of Time.

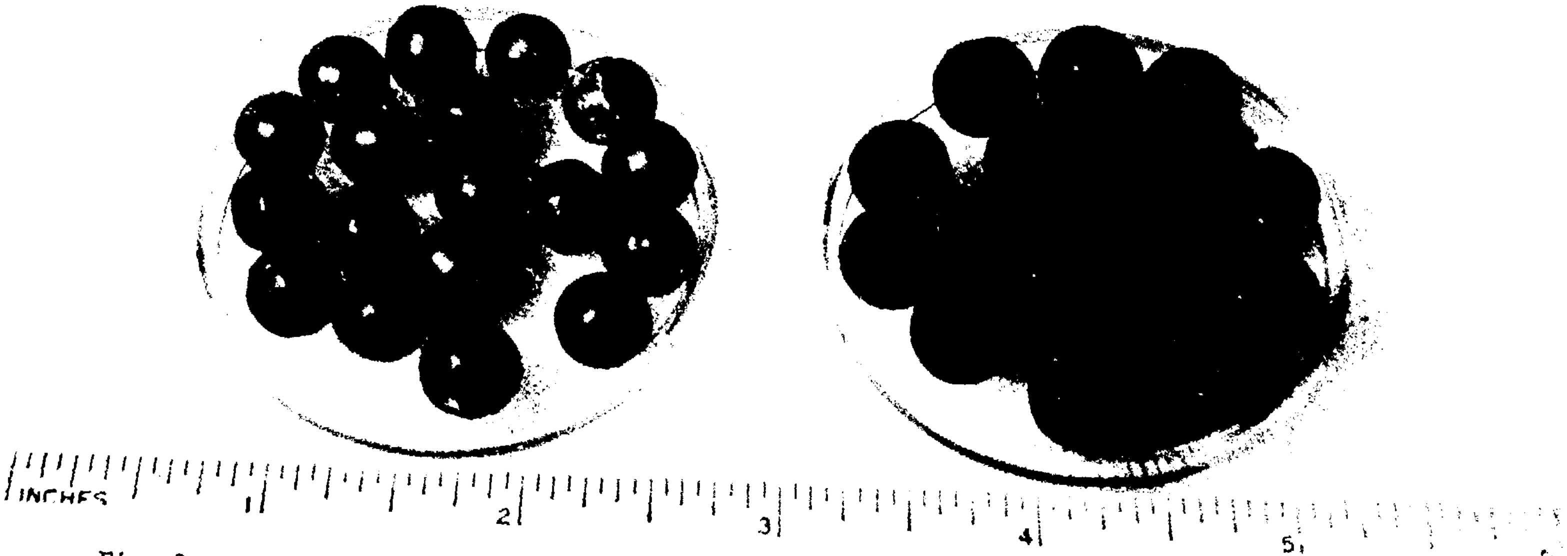


Fig. 3. Examples of Some Graphite Covered Sm_2O_3-Alumina (left) and B_4C-carbon (right) Balls.

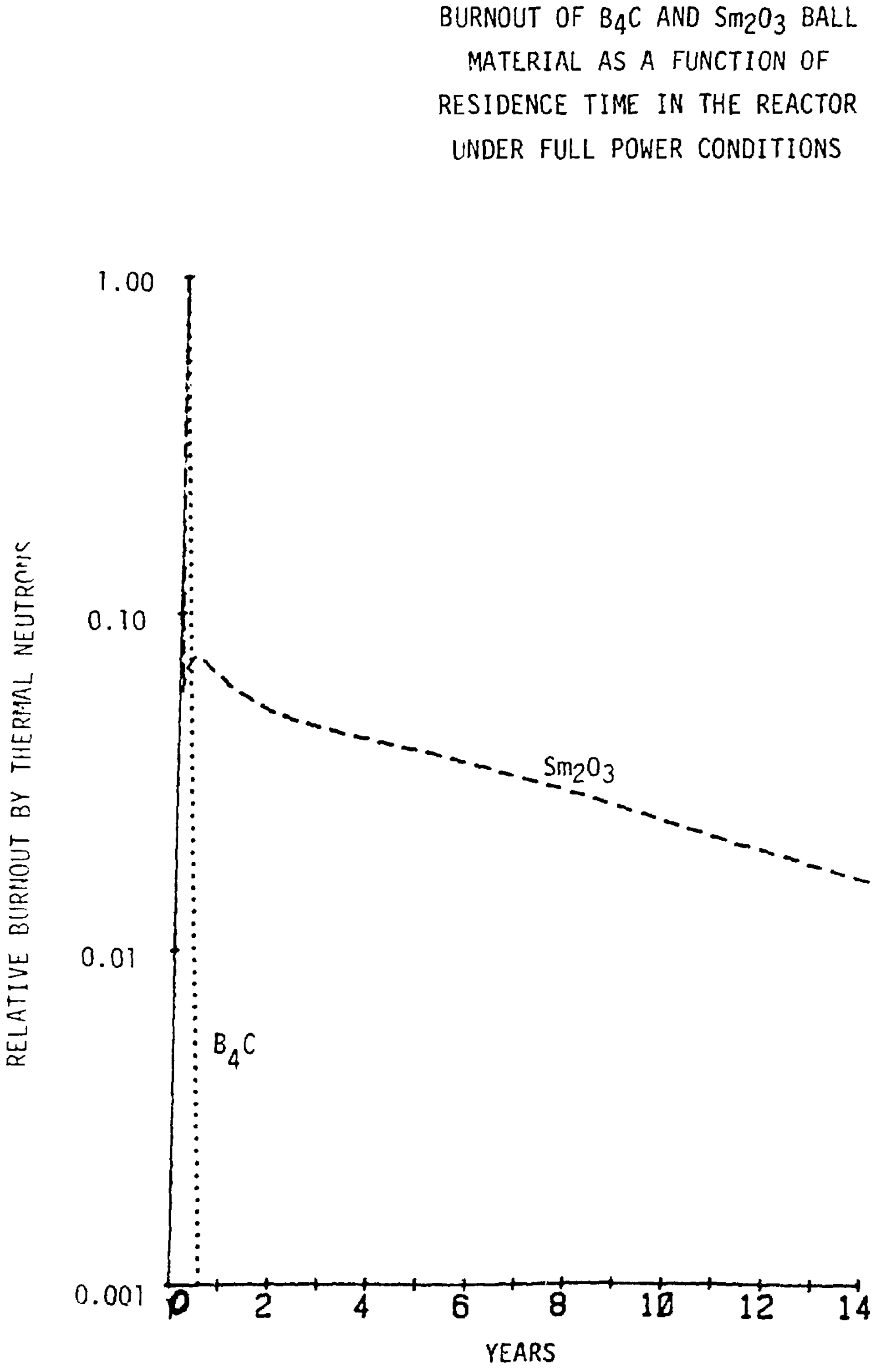
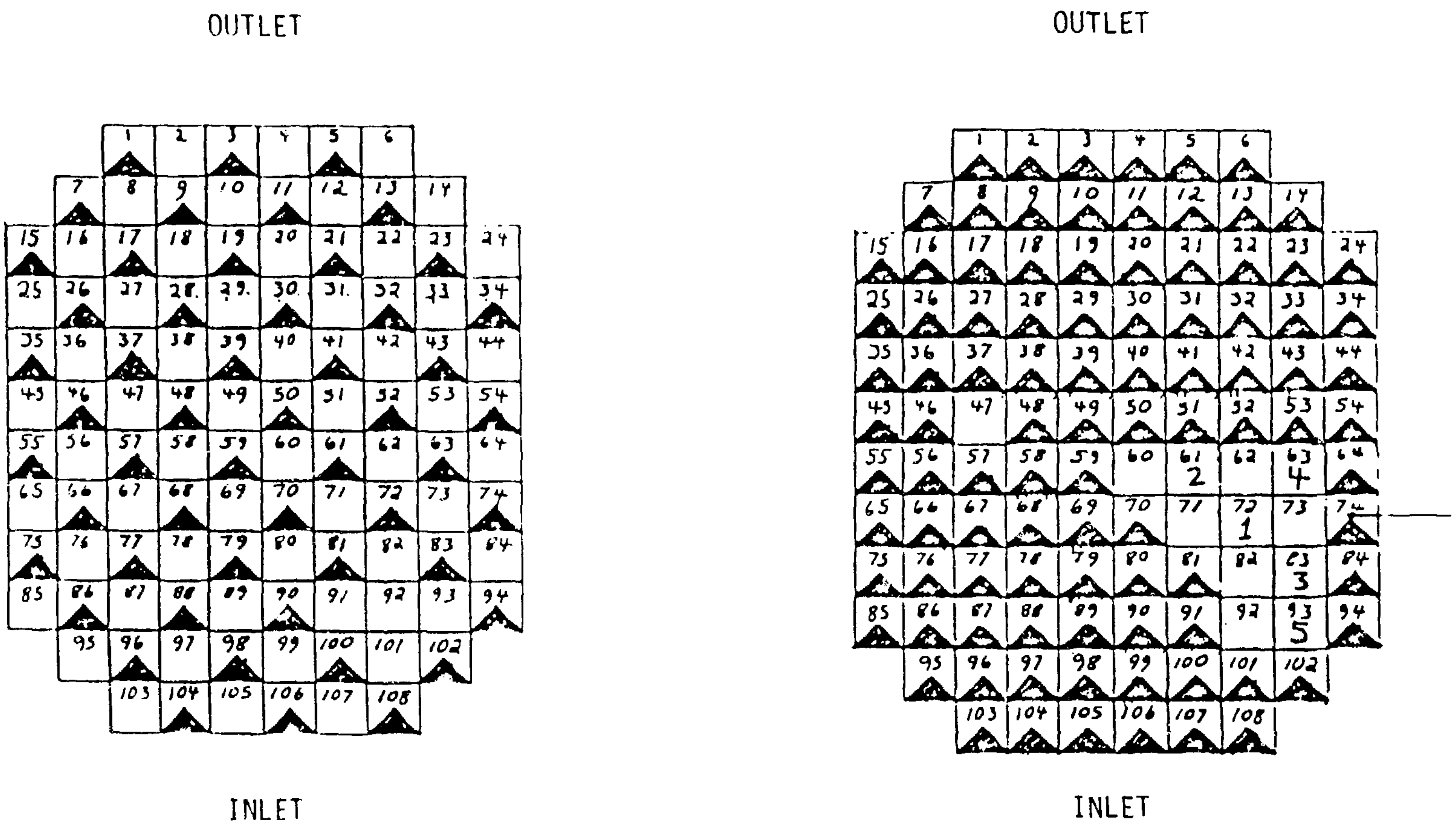


FIGURE 4

Fig. 4. Burnout of B_4C and Sm_2O_3 Ball Material as a Function of Residence Time in the Reactor Under Full Power Conditions.



UNIFORM BALL PATTERN

MINIMUM BALL PATTERN

BALL CHANNEL LOADING CONFIGURATION AT CRITICAL

FIGURE 5

Dup

A PRELIMINARY REVIEW OF BEYOND-DBA PWR ACCIDENT SEQUENCES

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ABSTRACT

Attempts to identify likely PWR accidents that are beyond the design-basis-accidents are based on US commercial operating experience and WASH-1400 results. Four sequences are described which involve combinations of failures, each of which has occurred at a US PWR at least as a single failure. The sequences are pressurizer-valve LOCA, interfacing-systems LOCA, complete station blackout, and total loss of feedwater. It is important to anticipate the occurrence of these accidents in PWRs so that operators can identify and deal effectively with them.

INTRODUCTION

This summarizes recent work to identify the most likely beyond-design-basis-accident (DBA) sequences which could occur in US commercial pressurized water reactors (PWRs). These are defined here as postulated sequences in which all events have been observed separately in US reactors. They involve an initiating failure and one or two additional failures. These sequences are deemed more likely than others in which all failure events have not been observed or which involve more failure events. The postulated accident sequences are taken from the larger set contained in the Reactor Safety Study (WASH-1400).[1] Reactor operating experience is from summaries of licensee event reports (LERs).[2-4]

Engineered safety features such as the emergency core cooling system significantly reduce public risk from most accident initiators and sequences. As a result, the reactor accident risk tends to be dominated by accidents that in some way bypass the engineered safety systems. These sequences are generally of extremely low probability, but some involve high consequences. For example, 99% of the early fatality risk from postulated PWR accidents is due to the three most severe of the nine WASH-1400 reference categories of radioactivity release.[1] Despite this, the most probable (although not necessarily of highest risk) beyond-DBA sequences are very unlikely to involve severe public health consequences. Instead, severe damage to the reactor, hardship to the utility and its customers, and intense public fear of the perceived threats to health and safety are more likely outcomes of reactor accidents than are the high-consequence outcomes that dominate risk. Study of the more probable outcomes of accidents may have high pay-off in terms of public health and safety as well as public confidence and continued energy availability. It also can provide a more realistic basis both for emergency planning and operations.

Four of the most likely beyond-DBAs are discussed below. Footnotes are included in the descriptions, and refer to actual US commercial PWR experiences.

PRESSURIZER-VALVE LOCA

A pressurizer-valve loss-of-coolant-accident (PV-LOCA) can result from two successive failure events. The first is a failure of some reactor protection system to prevent primary system pressure from reaching the opening setpoint for relief or safety valves following any steam-side transient. The second is failure of the open valve to close or reset following pressure relief. The result of these failures is a small loss of primary coolant through the stuck pressurizer-valve.

This is similar to the Three Mile Island (TMI) accident, except that the first event that caused the pilot-operated relief valve (PORV) to open was not a failure at TMI. The pre-TMI design of Babcock and Wilcox (B&W) reactors normally required the PORV to open following a steam-side transient. Current PWR designs normally require delayed reactor trip or delayed auxiliary feedwater to cause a sufficient pressure surge to open relief or safety valves.

US commercial reactor experience demonstrates the susceptibility to pressurizer-valve LOCAs. Unlike the single-failure sequences that occurred at three B&W reactors (Three Mile Island-2, 1979; Davis-Besse-1, 1977; Oconee-3, 1975),[a] no double-failure PV-LOCAs have been observed to date. However, the single-failures involved in PV-LOCAs have been observed separately.[a-d]

Analogous sequences in WASH-1400 are "TMLQ" (including auxiliary feedwater failure) and "TKQ" (including scram failure). According to WASH-1400, each has a frequency of occurrence of 10^{-6} per reactor-year. This figure varies among operating PWRs, because of differences in the reliability of auxiliary feedwater systems. Further, it is likely to be low because it is based on one failure of valve closure per 100 events, which is probably optimistic given the B&W experience.[a]

INTERFACING-SYSTEM LOCA

An interfacing-system loss-of-coolant-accident (IS-LOCA) can result from a breach of an interface separating the high-pressure primary system of a PWR from an appended, low-pressure system, such as the accumulators or the residual heat removal system. Since two check valves or two closed gate valves in series usually form the interface, double-failure is usually required to cause an IS-LOCA. The result of an IS-LOCA could be loss of primary coolant through a relief valve or an overpressure-ruptured appended system.

At least five incidents of spontaneous back-leakage through check-valves and into accumulators have occurred at US commercial reactors (R. E. Ginna, 1974; H. B. Robinson-2, 1976; Surry-1, 1976; Zion-1, 1976; and Arkansas One-2, 1979).[e] Further, at least one case of failure to close an accumulator isolation valve has been observed (Zion-1, 1976),[f] although not during an

IS-LOCA. In each case of check valve back-leakage to date, serious overpressure was avoided because of operator action to relieve accummulator pressure or because back-leakage was only momentary.

The IS-LOCA sequence in WASH-1400 is designated "V". The estimated probability of occurrence is given in WASH-1400 as 4 X 10^{-6} per reactor-year. This figure varies among PWRs, because of different valve combinations and different maintenance schedules.

COMPLETE STATION BLACKOUT

Complete station blackout (CSB) can occur if off-site AC power to run the plant is lost, and if emergency on-site sources of AC power are not available. In this situation, coolant injection pumps would be inoperable, and secondary steam relief would be complicated because of the unavailability of component cooling for the turbine-driven auxiliary feedwater pump. Loss of off-site power can be caused by a variety of single component failures. Unavailability of emergency on-site power requires the independent failure of each diesel generator, or common-mode failure of all diesel generators.

Off-site power loss occurs regularly in commercial PWRs due to numerous causes,[g] and is normally accommodated with plant safety systems. Similarly, numerous causes of diesel generator failures have been discovered both during testing and during loss of off-site power events.[h] In every loss of off-site power event to date, at least one of the redundant diesel generators was operable, although in at least one case all but one generator failed.[i]

A further complication of a CSB event would be additional unavailability of emergency DC power from batteries which power valve actuation motors and instruments. At least one incident of a battery fire has occurred after applying a load to the batteries.[j]

The WASH-1400 sequence, "TMLB'" is analogous to CSB. The estimated probability of occurrence is 3 X 10^{-6} per reactor-year. This probability varies among reactors, primarily because of different numbers of redundant diesel generators.

TOTAL LOSS OF FEEDWATER

Total loss of feedwater (TLOFW) can result from auxiliary feedwater unavailability following main feedwater trip. During a TLOFW no water would be provided to the steam generators for cooling of the primary system. This situation is similar to the pressurizer-valve LOCA described earlier, except that relief and/or safety valves would be forced to remain open because of high system pressure in a TLOFW event. Main feedwater trip is a normal transient event in commercial PWRs and can result from any one of a variety of single failures.[a] Auxiliary feedwater unavailability requires common mode failure or a series of independent failures in the auxiliary feedwater piping, pumps, or valves.

At Three Mile Island, the three auxiliary feedwater pumps on Unit 2 were valved-out, preventing auxiliary feedwater from reaching the steam generators.[b] In this case, however, the system was recovered after eight minutes, and did not affect the course of the accident, other than to cause confusion. Numerous incidents of single-failures have occurred in commercial PWRs,[k] and there has been at least one situation in which all but one auxiliary feedwater pump did not function.[l]

TLOFW is designated as "TML" in WASH-1400, and the frequency of occurrence is estimated to be 6 X 10^{-6} per reactor-year. The chance of total auxiliary feedwater unavailability is significantly less here than for the related pressurizer-valve LOCA discussed earlier, because in the TLOFW case there is some time available to the plant personnel to recover auxiliary feedwater. The probability of a TLOFW event varies among existing PWRs, because of design differences in auxiliary feedwater systems (and steam generator inventories). Some designs include significantly greater redundancy. Further, the consequences of a LOFW vary significantly among PWRs, because some reactors do not have the capability to inject emergency core cooling water at a high rate against the high primary pressure expected during a TLOFW. The latter tend to have highly redundant auxiliary feedwater systems (e.g., R. E. Ginna Nuclear Station).

CONCLUSIONS

Each of the events discussed in this paper (pressurizer-valve LOCA, interfacing-system LOCA, complete station blackout, and total loss of feedwater) go one step beyond the design-basis-accidents, because each involves one additional equipment failure. None are newly discovered potential accidents, because they were identified in WASH-1400, but they have only recently become worthy of careful consideration with the new interest of the nuclear industry in multiple failures. Further, the reactor operating experience which has tripled since the writing of WASH-1400 demonstrates that these beyond-design-basis accidents are not incredible.

It is particularly important to anticipate these potential accidents in US commercial PWRs, because the Three Mile Island accident showed that operator response is uncertain when faced with unanticipated situations. Clearly, if the beyond-design-basis-accidents discussed here become anticipated accidents at every PWR, although not necessarily design-basis-accidents, the contribution to reactor accident risk from these causes will be substantially reduced.

FOOTNOTES

a A relief valve stuck open in Three Mile Island-2, after opening in response to high pressure in the primary system. Similar events, although at significantly lower reactor power, occurred at Davis-Besse-1 in 1977, and Oconee-3 in 1975. These involved stuck-open pressurizer relief valves which were not immediately noticed by operators.

b Loss-of-heat-sink events have occurred at Three Mile Island-2, in 1979, and at Rancho Seco in 1978. The main feedwater pumps were turned off by the integrated control system (ICS) following loss of non-nuclear instrumentation at Rancho Seco after a light bulb was inadvertently dropped in a circuit panel. The auxiliary feedwater pumps were also automatically valved out by the ICS. When main feedwater was lost at Three Mile Island, the auxiliary pumps were found to be inadvertently valved-out.

c Recent investigations of control elements in nuclear plants turned up a degradation problem in control rod guides. If the degradation becomes too severe, it can prevent safety rod insertion. Combustion Engineering reactors were found to be most susceptible to this degradation, although Westinghouse and Babcock & Wilcox are also susceptible.

d During a test at H. B. Robinson-2, a stuck reactor trip relay was found. At Oconee-2, a valve was inadvertently left closed, preventing a reactor trip relay from functioning during a test.

e Momentary back-leakage through accumulator check valves was detected at R. E. Ginna in 1974 and at Zion-1 in 1976. Slow back-leakage occurred at H. B. Robinson-2 in 1976; Surry-1 in 1976; and Arkansas One-2 in 1979.

f During cool down, an accumulator isolation valve could not be closed using the switch in the control room. This occurred at Zion-1 in 1976.

g An insulation failure in the main transformer of the Beaver Valley switchyard led to complete loss of off-site power. The crash of a light airplane near Indian Point-1 caused a loss of off-site power event in 1967. A blown fuse caused loss of power at Indian Point-1 in 1970. A faulty inverter caused loss of power at Turkey Point-3 in 1973.

h Diesel generator failures have been noted during tests. Dresden-2 diesels failed to operate due to engine overspeed. Davis-Besse diesels failed due to a logic error in the safety feature actuator system. D. C. Cook-2 diesels were inadvertently removed from service instead of the diesels for the shutdown Unit 1. A Surry-1 diesel failed due to a cracked cylinder head.

i Two of three diesel generators were considered inoperable during a test at Zion-1 in 1973, due to an oil cooler tank leak in one generator and a leaking o-ring on the air pressure control of the other.

j Loose battery connections at H. B. Robinson-2 caused overheating and a battery fire when a load was applied during a test.

k Bearing failure discovered during surveillance testing rendered an auxiliary feedwater pump inoperable at Arkansas One-1 in 1977. Damaged suction line for an auxiliary feedwater pump was discovered at Calvert Cliffs-2 in 1977. Vapor binding prevented restart of a turbine-driven auxiliary feedwater pump at Beaver Valley[1] in 1976.

1 Two of three auxiliary feedwater pumps did not function at Zion-1 in 1977 due to failure to start the turbine-driven pump and inoperable service water lines to one motor-driven pump.

ACKNOWLEDGEMENTS

This work was performed under the auspices of the US Nuclear Regulatory Commission.

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SESSION XVIII

HUMAN FACTORS IN NUCLEAR POWER PLANT OPERATION

Chairmen

W. Bastl - University of Munich

H. G. O'Brien - Tennessee Valley Authority

HUMAN FACTORS IN NUCLEAR POWER PLANT OPERATION

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ABSTRACT

More than 30% of the events reviewed in the Licensee Event Reports (LER's) involve human factors in the sense that human perception, decision-making, forgetting, commission of an incorrect response, or omission of a correct response were involved. Comparatively little attention has been given to the study of human reliability, despite its importance in the operation of nuclear power plants. Thus, an extensive effort is being devoted to developing a comprehensive human factor program that encompasses establishment of a data base for human error prediction using past operation experience in commercial nuclear power plants. Some of the main results of such an effort are reported including data retrieval and classification systems which have been developed to assist in estimation of operator error rates.

Also, statistical methods are developed to relate operator error data to reactor type, age, and specific technical design features. Results reported in this paper are based on an analysis of LER's covering a six-year period for LWR's. Developments presently include a computer data management program, statistical model, and detailed error taxonomy.

INTRODUCTION

Events at the Brown Ferry and Three Mile Island No. 2 plants, as well as numerous less severe incidents, have underscored the importance of human reliability but have also warned of its sensitivity to unforeseen sequences of events and to error-induced stress. Comparatively little attention has been given to the study of human reliability, despite its importance in the operation of nuclear power plants.

A multidisciplinary team of nuclear engineers, applied statisticians, industrial psychologists, and human factors engineering experts was established at Iowa State University, Ames, Iowa in 1975, and thereafter jointly with Science Applications, Inc., to evaluate the impacts of human factors on nuclear plants safety, availability and reliability. Figure 1 illustrates the approach taken by the team.

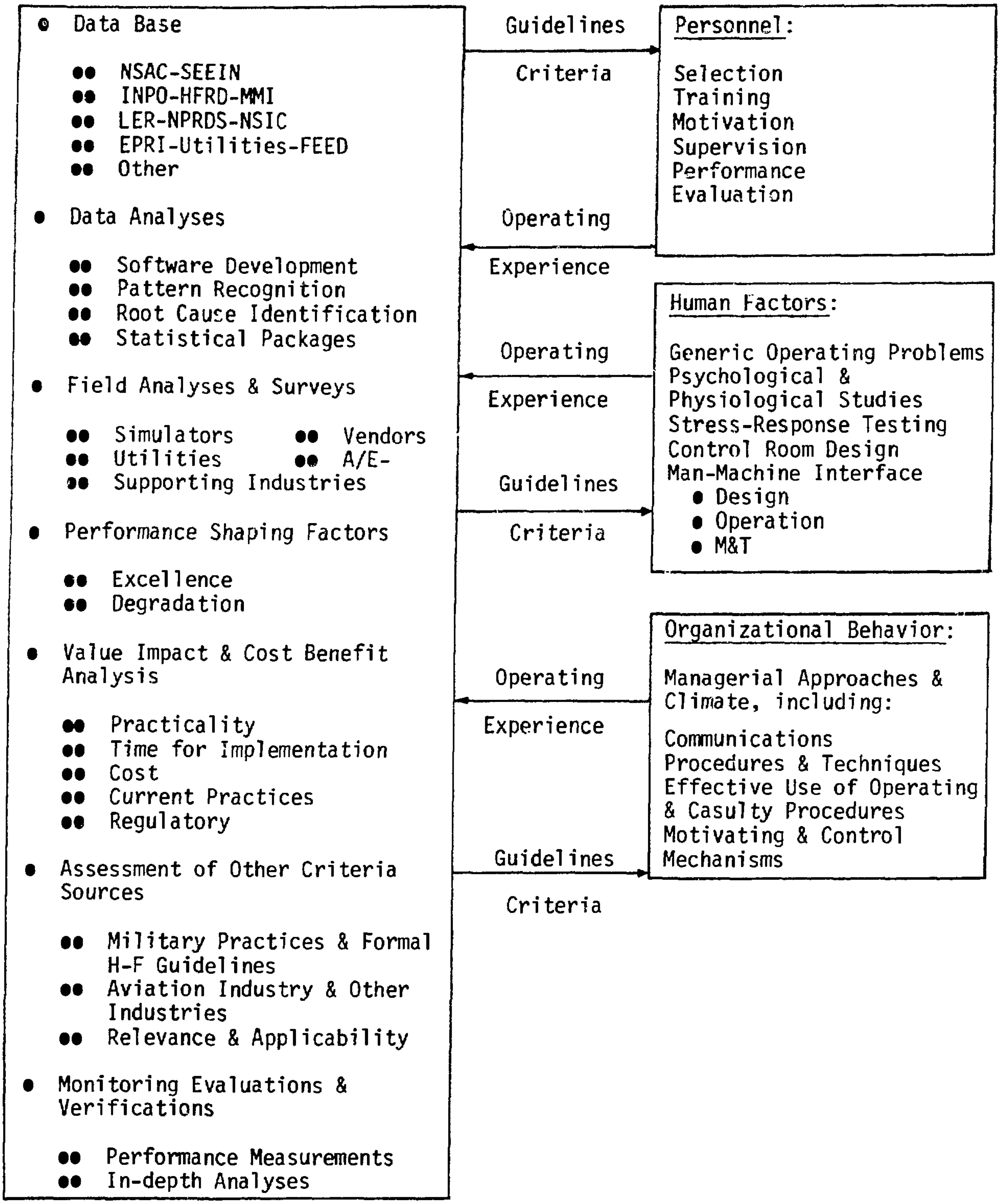

Figure 1. Comprehensive Integrated Human Factors Program

The LWR operation experience documented in available data sources is analyzed utilizing appropriate software and statistical packages to identify generic and plant specific problems. Field analysis and surveys are conducted to evalute the impacts of current practices related to personnel selection and training, and organizational behavior on plant performance. Where modifications are needed to current practices as it relates, for example, to control room design or maintenance and testing and operation practices or to hardware and plant layout or component designs, these recommended modifications are evaluated for their practicality, time for implementation, cost and licensability.

Criteria and guidelines developed and utilized regularly in other industries such as the military and aviation industry are assessed in terms of their relevance and applicability to the nuclear industry. Evaluation of current practices in the nuclear industry indicate an under utilization of the well-established criteria and guidelines related to human factors engineering in other industries.

As part of the data analysis and monitoring subprogram, a data base for human error prediction was developed based on past operation experience in commercial nuclear power plants. Also, data retrieval and classification systems were developed to assist in estimation of operator error rates. The LER's in their present forms do not provide a taxonomy or quantification of errors needed to relate events to design data, operations management, training, or human engineering. [1,2] It is also important to develop methods of relating operator error data to reactor type, age, and specific technical design features.[3,4] Results reported in this paper are based on the analysis of LER's covering a six-year period (1972-1978) for Light Water Reactors (LWR's). The analysis makes use of computer data management programs,[4,10] statistical models,[5,6] and detailed error taxonomy [4,7] which have been developed specifically for this study.

HUMAN FACTORS IN THE CONTROL ROOM

In a previously published study [8,9] numerous design aspects of nuclear power plant controls and displays were discussed from a human factors standpoint. Design deficiencies associated with the control room included: excessive control and display panel size, L-shaped configurations, and excessive viewing and walking distances. On the control board itself, problems included: a lack of functionally demarcated panel areas, a separation of associated panel elements, and mirror-image location coding (simultaneous position coding to the right and left of a centerline for given sequence of displays or controls). Excessive read distances were often required, some necessitating climbing. Many controls were also identical in appearance. Numerous additional details were noted with respect to scale markings, parallax, glare, illumination, chart recorders, and limited communication capability of annunciators as used in the plants observed. Error inducing conditions included the absence of attention-demanding indicators to inform of malfunctions accurately, valves not lining up in ways that communicate their status visually, a general lack of control coding, and inadequate turnover in duty hours. Human factors in control system design involves more than hardware, but some of the errors logged in the LER's were related to the problems noted in the study just cited.

Examination of reported maintenance and testing (M&T) problems has shown the lack of coordination between the M&T crew and the control room operators and among different shifts in the plant. Frequent problems resulted from the use of the wrong work package and inadequate labeling of equipment.

OPERATOR-ANALYST STATISTICAL INFORMATION SYSTEM (OASIS)

In order to generate statistical trends, event frequencies and event criticalities from compiled data, a standard, comprehensive and accurate event classifications were needed. The OASIS model acts as an interface between plant operators, encoders, computer programers and data analysts.[4] The model:

1. Classifies events, actions, casualties, and phenomena associated with plant operation and nuclear behavior.
2. Provides one-to-one mapping between actual events and a condensed, coded format with minimal loss in factual information.
3. Facilitates computer manipulations of data parameters and extraction of particular events.
4. Covers all types of phenomena in a variety of plant types and operating conditions.
5. Is specific in describing phenomena which occur frequently or which have great impact on plant safety, reliability or performance.
6. Is flexible to permit expansion and new types of data and to accommodate additional plants or new design concepts.

An example of a procedural error described using the system is as follows: 50-029·R·68315·XXXXX·EGR·0100·I-AA·PX·ST·M·EF·PU·NNNN-0000·X·procedure error. This string decodes to yield: Plant docket 50-29, Yankee Rowe, routine report, 315th day of 1968, event data unknown, involving an electrical generator, slightly significant to machinery and operations, caused by improper action, discovered during scheduled testing by a maintenance man; correct action followed in response, resulting in unchanged plant status, no safety hazard and no damages, event frequency unknown.

GENERAL EVENT CLASSIFICATION SYSTEM (GENCLASS)

GENCLASS was developed to provide an alternative system to OASIS and to provide a different method of counting operator errors.[4] In GENCLASS, provision is made to include human errors in activities other than operations and also to account for system and component failures. The coding sheet for GENCLASS is shown in Figure 2.

HUMAN-ERROR TAXONOMY

Facility　Reactor Type　Power Level Mw(t)　Name　Date

1	2	3	4	5	6	7	8	9	10	11		13						14	15	16	17	18		19		
												Consequences														
																Safety										
Reference Number	Docket Number	Ck. Digit	Serial Number	Date of Report	Date of Event	Punched Date	Phase of Operation	Outage	Error Domain	Human Act	Taxonomy	Equip.	Effect Task	Effect Operation	Duration	Rad Release Environ	Rad. Exposure	Human Causes	Human Error Type	Oper Stress	System	Component	Location	Critical	Code	Comments
1 2 3 4 5 6	7 8 9 10 11	12	13 14 15 16	17 18 19 20 21 22	23 24 25 26 27 28	29 30 31 32 33 34	35 36 37 38	39 40	41 42 43 44	45	46	47 48	49 50	51 52 53	54 55 56 57	58 59 60 61	62 63 64 65	66 67	68 69 70 71	72 73 74	75	76 77	78	79	80	

Figure 2. Worksheet for recording events for GENCLASS

Examples of information groupings in GENCLASS are: outage duration, type of error (human activity involved), mode of error, task taxonomy, consequences, error cause, error type, stresses, system involved, location and criticality.

Mode of error includes three possible classifications of statistical occurrence: systematic (small dispersions about a norm), sporadic (small dispersion about a norm with an occasional outlier), and random (large dispersion). Task taxonomy includes classification as: simple, complex, vigilance, control, emergency, abnormal, and operator incapacity. Not all operator errors affect the completion of operation tasks or cause component failures. The most frequently occurring operator errors do not result in equipment malfunctions. In terms of impact of operator error in the plant system and components, operator errors are divided into: (a) non-malfunction-producing operator errors (NMOE) and (b) operator-induced failure (OIF). NMOE's are "reversible" in the sense that they cause delay, but do not induce malfunction and repair. A high frequency of NMOE's suggests a likely OIF on a given task. An OIF is irreversible. It may affect components only and not affect plant system performance because of component redundancy. Some causes of operator errors are: misunderstanding of procedures, communication problems, use of incorrect procedure, lack of coordination with maintenance, misidentification of alarm, inadequate control room layout (inadvertent activation of controls, misreading of instruments), lack of guidelines, misunderstanding of technical specifications, disregard of procedures, checklist not completed, procedural deficiency. Figure 3 gives a sample of error percentages as fractions of total operator error for various components. The importance of human engineering for valves and switches is clearly evident.

LER RETRIEVAL SYSTEMS

A computerized system was designed for retrieving LER data as originally reported and as coded and classified by GENCLASS for purposes of error analysis. 4 Data is stored on disks and tapes. Keywords are used to retrieve selected information. The system, known as LERRETS, consists of three phases:

Phase I:	Storage of data into source file, master file, and accumulated movable file. Old data can be updated.
Phase II:	Separation of citations and keyword files from sequential source LER.
Phase III:	Keyword index, citation index, and user request are used to select specific documents.

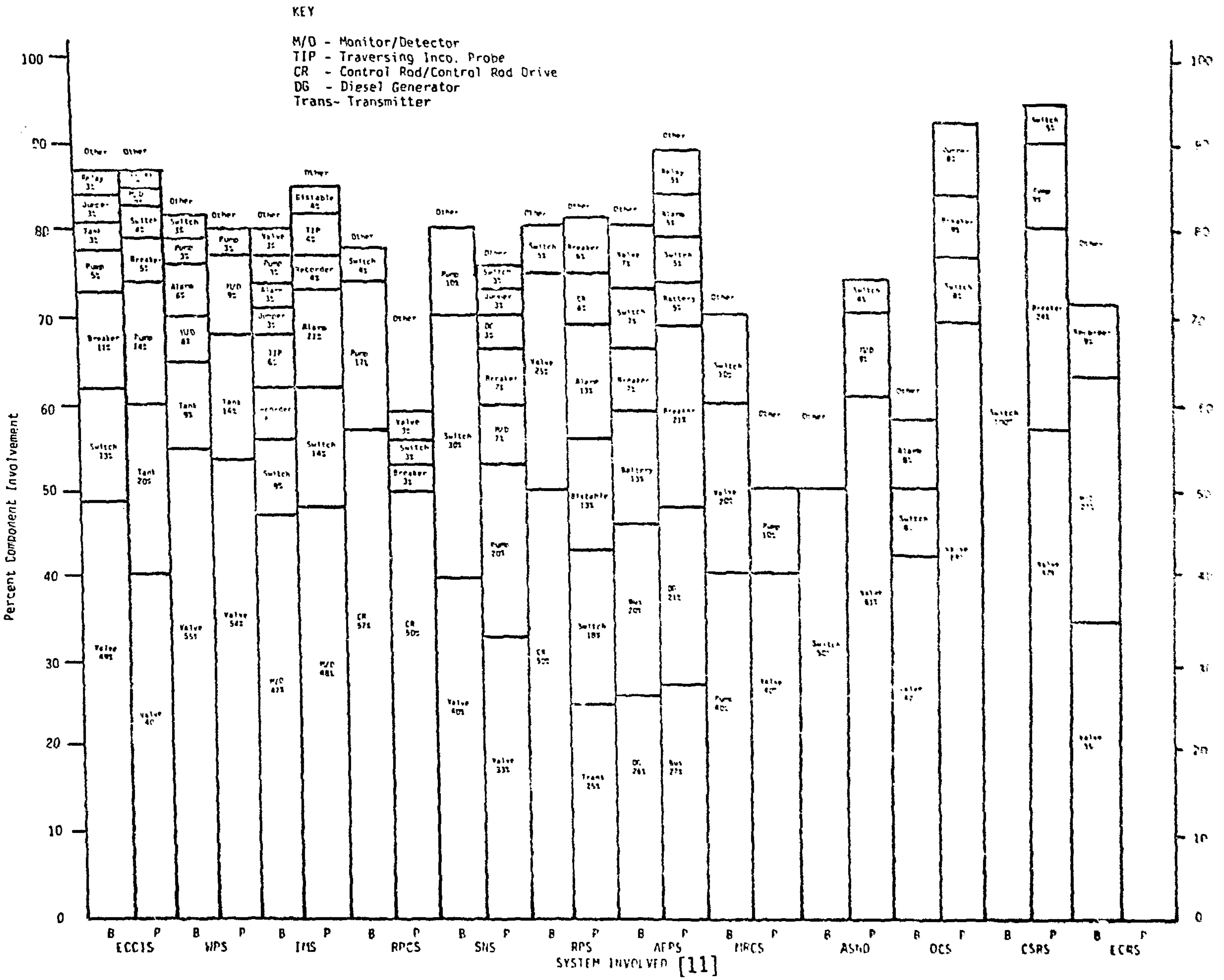

Figure 3. System delineation with the most frequently involved components in operator errors

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THE RELATION OF MANAGEMENT, SUPERVISION, AND PERSONNEL PRACTICES TO NUCLEAR POWER PLANT SAFETY

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ABSTRACT

The knowledge base of industrial/organization psychology suggests three major areas of research with important implications for nuclear power plant safety. These areas are Management and Supervision; Personnel Selection, Training and Placement; and Organizational Climate.

Evidence drawn from several Three Mile Island investigations confirms that organizational structure of plants and supervisory practices, the selection and training of personnel, and organizational climate are important factors.

Difficulties in decision making and coordination of personnel are pinpointed. Deficiencies in training are highlighted and the climate of working atmosphere is discussed. These matters are related to nuclear power plant safety. Future research directions are presented.

INTRODUCTION

The purpose of this paper is to assess areas of research in management, supervision, and personnel suggested by the knowledge base of industrial/organizational psychology and which hold promise for producing information with important implications for nuclear power plant safety. The research ideas have developed from research conducted by the Nuclear Safety Research Group (NSRG) of Iowa State University and Science Applications, Inc. - Ames (SAI-Ames) and by analysis of reports of investigations of the Three Mile Island (TMI-2) accident.

The Nuclear Safety Research Group (NSRG) and Science Applications Inc. - Ames (SAI-Ames) have been doing research on nuclear reactor power plant safety for several years under the direction of Drs. Zeinab Sabri and Abdo Husseiny. The authors, psychologists with research and managerial experience, have been members of the research team.

The basic data for our research have come from Licensee Event Reports (LERs), and other available Occurrences Information Sources (OIS). Our efforts have been focused primarily on human factors in systems design and in human machine interaction as they affect safety of nuclear power plants. We have made a special study of the impact of maintenance and testing activities on plant safety [1].

One can infer from reports of various safety related events in plants that management, supervisory and personnel practices play a very important part in determining safety. This inference derived from the LER analysis is confirmed by the reports from investigations into the Three Mile Island accident. First, some results from our research will be presented and then reference will be made to various reports of investigations into the Three Mile Island accident.

Here are a few examples of human "errors" gleaned from Licensee Event Reports.

- A maintenance worker with a work order to repair a defective valve removed the internal mechanism of the valve, put the cover back, and marked the valve repaired.

- An operator left the control room to make a personal telephone call. This decreased the number of operators in the control room below that required by regulation.

- Painters put paper and maksing tape over the exhause vents of the containment building as they painted the inside of the building. They neglected to remove the paper and tape after they finished painting. The blocked vents were discovered when an operator tried to exhaust air.

- During refueling a control room operator who was supposed to monitor neutron flux level neglected it because of poor communications and too many demands on his time. The flux level went out of bounds and some workmen received abnormal amounts of ratiation.

On the surface these "errors" can be considered simply that, errors of omission or commission, reflecting inadequacies on the part of the persons involved; but viewed from the perspective of psychology these "errors" occurred because of inadequacies in management and supervision and an organizational climate conducive to error making.

One of the inferences from research on the impact of maintenance and testing on plant safety supports the above perspective.

The most frequent types of errors, which were major contributors to the overall maintenance and testing error rates, were coordination errors between a test station and the control room. For example, a technician places an instrument in test as scheduled. The control room operator for some reason or other, is unaware of the testing activity. Testing activities set off control room alarms and the operator takes inappropriate actions which affect plant operability.

Further information from the maintenance and testing study reflects the impact of management.

An investigation of nine plants revealed a negative correlation between the fraction of the utility's operating expenses applied toward maintenance and the average number of maintenance and testing events reported in the (LER's) for each plant owned by the utility. The average number of maintenance and testing events per plant for each utility decreases as the ratio of maintenance expenses to operating expenses increased. We also found that the average number of maintenance and testing events/plant increases as the return on each dollar invested increases. (This finding is based on the utility as a whole and not the individual plant.)

Obviously, management decisions have an impact of the quality of maintenance and testing operations.

Now let us turn to reports on the TMI accident.

Utility companies have tended to be quite conservative in their management and personnel practices. It is probable that the demands made by nuclear plants as compared to those made by fossil fuel plants require new approaches by management to ensure optimal operating efficiency and safety.

The report of the President's Commission on the Accident at Three Mile Island commented on this point. [2] "When the decision was made to make nuclear power available for the commercial generation of energy, it was placed into the hands of the existing electric utilities. Nuclear power requires management qualifications and attitudes of a very special character as well as an extensive support system of scientists and engineers. We feel that insufficient attention was paid to this by General Public Utilities (GPU)".

The Commission recommended ".... the development of higher standards of organization and management that a company must meet before it is granted a license to operate a nuclear power plant."

Some negative comments about management gleaned from the Commission reports are:

"The Met Ed management systems, procedures, and practices did not provide Met Ed, a firm understanding of TMI's operations, nor were effective systems of checks and balances in use."

"Utility management did not require attention to detail as a way of life at Three Mile Island." There were 14 specific examples of neglect of detail listed under this heading in the report.

The Commission recommended "Integration of management responsibility at all levels must be achieved consistently throughout this industry. Although there may not be a single optimal management structure for nuclear power plant operation, there must be a single accountable organization with the requisite expertise to take responsibility for the integrated management of the design, construction, operation, and emergency response functions and the organizational entities that carry them out. Without such demonstrated competence a power plant operating company should not qualify to receive an operating license."

The report of the Nuclear Regulatory Commission Special Inquiry Group [3] concluded, "The one theme that runs through the conclusions we have reached is that the principal deficiencies in commercial reactor safety today are not hardware problems, they are management problems," .... We have found based on our study of TMI and our interviews with knowledgeable people in the industry, that many nuclear power plants are probably operated by management that has failed to make certain that enough properly trained operators and qualified engineers are available on site in responsible positions to diagnose and cope with a potentially serious accident."

The Special Inquiry Group also stated, "...the variation in nuclear capability among the various utilities appears to be that different utility companies accord their nuclear generation units different priorities and different amounts of resources." and

"The fact remains that nuclear technology is different in kind from the traditional technology of electric generation by fossil fuel and hydroelectric means--more dangerous, more sophisticated and more demanding of advanced management, maintenance, and quality control. It may be that some utilities, because of their limited size, limited technical staffs or limited capital simply will not be able to meet the increased demands we think the Three Mile Island accident demonstrates must be made upon them by the NRC to provide a technically competent site management team on every shift, first class operator training programs and other safety improvements."

As we prepared this paper and mused on management problems in utilities we were struck with a somewhat irreverant question, "What would have happened to our space program if it had been entrusted to the railroads?"

Industrial/Organization psychology provides us with a theoretical framework and some constructs (concepts) which help us to think about organizations and their management in a way which can increase our understanding of the relationship of management to nuclear power plant safety.

Let us begin by considering in historical focus some ways of conceptualizing organizations.

There are a number of views or perspectives on organziations but we will discuss briefly only the three which are the most prominent schools of organizational theory. They are in terms of historical development: the classical, the human relations, and the systems approaches.

CLASSICAL/BUREAUCRATIC THEORY

This theory which grew out of the Industrial Revolution uses scientific and machine-like concepts to view organizations. Organizational effectiveness is thought to come from a rigid centralized, pyramid of authority structure in which the number of hierachical levels, the spans of control and line-staff relations are carefully delineated.

Bureaucracy consists of clearly specifying the work to be done, clearly defining each job and its part within the complex of rules and regulations and of selecting and training persons with the appropriate characteristics to operate the system.

Thus, the classical model views effective operation as depending upon coordination and specialization being built into the structure of an organization. Other principles associated with classical organizational structure are: the prime principle, unity of command; span of control; the exception principle; departmentalization; line-staff distribution of function, and the profit-center concept. In terms of its assumptions about human behavior in organizations, the classical approach assumes that the individual will always act in the best interests of the firm, is motivated primarily by financial incentives, and can be relied upon to act rationally and mechanically.

THE HUMAN RELATIONS VIEW

The human relations school developed as a reaction to the emphasis upon the machine-like characteristics of men and organizations implied in classical theory. Far from being a cog in the machine, the individual is of central concern. There is a deep concern for attitudes, values, and emotional responses. There is a focusing on the ordering of relationships but it emphasizes persons instead of positions. Control of behavior is viewed as residing within the individual rather than determined from without. The human relations school emphasizes the concept of basic needs which are organized in hierarchical levels of importance with survival needs such as food and shelter at the bottom and self-actualization (self-development) at the top. Organizations are to be structured with an emphasis upon freedom, democracy, and the dignity of the individual. The organization will be most productive when organization members can satisfy their social, ego, and self-actualization needs. The most effective organization permits individual antomony and thus maximizes task involvement and motivation from within.

There is an emphasis on group dynamics, the "informal organization", and management syle which is employee-oriented.

SYSTEMS THEORY

System means a set of interdependent, interacting elements, a group of units combined together in an organized form so that a change in any one part affects the other. An organization is a collection of interdependent sub-units, a set of interacting elements.

There are a number of properties of organizations as viewed by systems theorists. Input, throughput and output are processes associated with organizations, and major subsystems of organizations are a production or technical component; a production supportive component (purchasing, sales); a maintenance component (personnel, production maintenance); an adaptive component (research and development, organization and methods, market research); an institutional component whose function it is to obtain social support and legitimacy for the organization; and a managerial component that coordinates internal and external activities and resolves conflicts. A change in the forces shaping an organizational system (social, technical, or economic) has repercussions for all parts of the system. Similarly, a change in any one of the variables such as the task, the structure, the technology, or the "people" of the organization can set off changes in the other three.

The systems approach calls attention to a wide range of social, psychological, economic, and technical forces operating on and within an organization. It emphasizes dynamic interrelationships. Organizations are made up of many diverse individuals and groups, multiple coalitions and alliances each behaving in a way to achieve its own goals and objectives. Each organization may be viewed as a more or less complex arena for internal bargaining among the bureaucratic elements and personalities comprising it. Its action is the product of their interactions.

Our perception is that utilities and particularly nuclear power plants can be classified as reflecting a comination of the Classical and Human Relations approaches. We are suggesting that improvements in nuclear power plant safety could be made through systematic research using Systems Theory as a guide. The research areas to be listed reflect primarily the systems approach. Some of these ideas have been presented as suggestions to the Institute for Nuclear Power Operations (INPO).

There are three major areas of investigation and research in management and personnel practices that should yield significant payoffs. These are: Management and Supervision; Personnel Selection, Training and Placement; and Organizational Climate.

MANAGEMENT AND SUPERVISORY PRACTICES

This area is perhaps the most important area for study because it affects all other areas. First of all, the organizational structure of plants should be studied. It appears that they tend to be "vertically" structured with most decisions made at the top. Alternative structures should be investigated, particularly those utilized under emergency conditions. Other areas of investigation are reflected in these questions.

- Who makes the decisions and how?
- How are the communciations and working relations structured for the managerial, the operations, technical and maintenance groups?
- What are the formal and informal communications channels?
- How is supervision organized?
- What are the management control mechanisms, the checks and balances?
- What are the mechanisms used to interrelate the several work groups?
- What reward systems are utilized?
- What is the role of the unions?
- What mechanisms are used to deal with stress and conflict resolution?

These are only a few of the questions that can be asked about management and supervision. The efficient operation and safety of plants depend on the best possible management.

SELECTION, TRAINING AND PLACEMENT OF PERSONNEL

This area of study requires both short and long range attention. A first step would be a survey of existing selection, placement and training practices as they pertain to plant managers, operators and operational personnel, technical personnel and maintenance personnel. The survey results should reveal present practices and areas needing strengthening. Noteworthy is TVA's attention to the testing of operators. But focus of attention on operators alone will only partially help ensure safety. Other personnel and especially managerial personnel must also receive attention.

ORGANIZATIONAL CLIMATE

Managerial, operations, technical and maintenance personnel need to work harmoniously and in close cooperation to ensure effective and safe plant operation. Organizational climate is a concept which reflects the working atmosphere or culture of an organization. It refers to the extent to which an organization provides a psychologically meaningful environmental setting which limits and influences workers' behavior as individuals and as groups.

Organizational climate can be viewed as a summary of perceptions members of the organization share about their work environments. These perceptions serve to guide appropriate and adaptive task behaviors. Based on a variety of cues present in their work environment, employees develop coherent sets of perceptions and expectations regarding behavior-outcome contingencies and behave accordingly.

To illustrate how the concept of climate derived from systems theory can be useful, consider the specific notion of safety climate.

Following are some characteristics of factories having successful safety programs. These characteristics as perceived by employees form the safety climate of that organization [4].

Overall, low accident plants have a strong management commitment to safety. In low accident plants (1) top management is personally involved in safety activities on a routine basis, (2) safety matters are given high priority in company meetings and production scheduling, (3) safety officers in safe plants have high status, (4) emphasis is placed on safety training, (5) there are open communication links and frequent contacts between workers and management, (6) there are frequent safety inspections, (7) there is generally good environmental control and good housekeeping, eg., there are orderly plant operations, controlled environmental conditions, and high usage of safety devices, (8) there is a stable work force with low turnover and older workers, (this finding may reflect good industrial relations programs and personnel development practices) (9) there are distinctive ways of promoting safety. (Guidance and counseling approaches rather than enforcement and admonition are used). Some other methods used are individual praise or recognization for safe performance and enlisting workers' families in safety promotions.

Quoting from Zohar [4], "When all these organizational characteristics are integrated, it is possible to form a coherent organizational pattern of a highly safe company". Management is actively involved in safety management and creates a general administrative control climate in which work is to be performed. This climate results in increased performance reliability of workers, good housekeeping, and high design and maintenance standards for work environments. There are well-developed personnel-selection training and development programs in which safe conduct is an integral part. Communication links between workers and management are kept open, enabling a flow of information regarding production as well as safety matters. Finally, general management philosophy is not strictly production oriented but also people oriented, as evidenced by various supportive policies described above.

Turning now to other research topics categorized under organizational climate and considered as questions.

- What are the several working climates in the plants in the utilities?
- Do the various groups comprising the organization work well together?
- Is there a feeling of team, that is, do the various work groups feel they are team members and work harmoniously?
- Is there conflict perceived amongst the groups?

- How is conflict resolved?
- How much interference and "sabotage" are present?
- What are the attitudes of plant personnel about working in nuclear plants?
- What is the level of stress and its effect on performance and safety?
- How much job or work involvement is there on the part of individual workers?
- How much and what kinds of communication occurs in a plant amongst the managerial, operations, technical, and maintenance groups?
- How much individual initiative and autonomy are allowed?
- How are workers involved in decision making?

The answers to these questions will help define nuclear organizational climate and allow correlation with plant safety.

This paper has drawn on industrial organization psychology, on research of NSRG and SAI-Ames, and reports from the Three Mile Island investigations to propose areas of research into Management and Supervision, Selection, Training, and Placement of Personnel, and Organizational Climate.

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ESTIMATION OF OPERATOR HAZARD FUNCTION BASED ON COMMERCIAL NUCLEAR POWER PLANT EXPERIENCE

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ABSTRACT

A statistical operator reliability function is synthesized for the operation of commercial nuclear power plants. The model is developed in a form appropriate for estimating and predicting a hazard function utilizing the Kalman filtering methodology. The methods of Least Square Fitting and Impulse Moment updating have been used for a priori estimator. Kalman filtering was used as a postriori estimator.

Data on operator events has been extracted from the Licensee Event Reports (LER's) for all U.S. Commercial Light Water Reactors from 1972-1977. The data was used to compute error rates which were then employed in the application of the model to examine effects due to age, power level, and reactor type on the operator hazard function.

INTRODUCTION

Present data on operator errors are limited and scattered, and hence smoothing and reliable estimation procedures are necessary for hazard function determination [1]. One such procedure is the use of the Kalman filter technique which has been demonstrated as a useful tool in many situations. The technique has been applied in different human factor studies; namely, learning during training [2] and performance of pilots [3]. The technique has also been proposed for prediction and updating of operator error rates in executing vigilance tasks [4].

The nuclear industry and utility companies in particular are concerned about the safe and economical operation of nuclear power plants which are directly affected by operator error. Since the TMI-2 incident, efforts have been made to improve operator performance. This necessitates the development of an operator error rate model or by estimating the hazard function for man-machine interactions. Once the hazard function for human errors is specified, several standard statistical reliability techniques, such as the generalized poisson or Markov processes, can be used.

Data on operator error rates has been extracted from LER's (Licensee Event Reports) for both Boiling Water Reactors (BWR's) and Pressurized Water

Reactors (PWR's). Proper collapsing and smoothing of the data has been done by taking into account plant availability. The effect of age, power, and the type of reactor on the operator error rate has been studied. To observe the static and dynamic behavior of learning parameters, these methods have been employed; namely, Least Square Fitting, Impulse Moment Updating, and Kalman Filtering Estimations.

THE MODEL

For a well trained operator, the rate of errors associated with a specific task is representd by [4]

$$\varepsilon(t) = \varepsilon_f + (\varepsilon_i - \varepsilon_f)\, e^{\alpha t} \tag{1}$$

provided operator performance is continuously monitored and procedures and regulations are revised accordingly and modified when necessary. Here $\varepsilon(t)$ is the operator error intensity or the instantaneous error rate; ε_i is the initial error rate at the start of specific task performance; ε_f is the final constant error rate.

$$\alpha = -\frac{1}{\tau} \tag{2}$$

The parameter τ represents a performance improvement time constant for the operator to achieve an error rate of ε_f after some time t. The error intensity should asymptotically approach the constant level

$$\varepsilon(t) = \varepsilon_f \tag{3}$$

for an experienced operator conducting a familiar task. The parameters ε_i, ε_f and α change in value as more data is updated. The exponential model has been verified using a data sample [5]. For the purpose of the present analysis, the hazard function $\lambda(t)$ may be expressed by rearranging Eq. (1) into the form

$$\lambda(t) = a(1 + be^{\alpha t}) \tag{4}$$

where

$$\lambda(t) = \varepsilon(t)$$

$$a = \varepsilon_f$$

$$b = (\varepsilon_i - \varepsilon_f)/\varepsilon_f\ . \tag{5}$$

SMOOTHING

Smoothing is achieved by two methods: window smoothing and integral smoothing. In window smoothing, the averaging process is done on (k-1) adjacent points. The smoothed data output can be defined as

$$U_0(ik'\theta) = 1/k \sum_{ik'}^{ik'+k} f(n\theta),$$

$$W = kT \text{ and } T = k'\theta \tag{6}$$

where θ is the selected time interval, T is the time delay for window sampling, W is the width of window, and k' is usually chosen to be unity to obtain the maximum number of smoothed data.

In integral smoothing, the integrating process at time t is done by considering all the points in the interval (0,t). The output of integral smoothing has the form of;

$$Z_m = \sum_{i=1}^{m} \lambda_i(t_{i+1} - t_i) - at_m \tag{7}$$

where λ_i is the number of errors observed in time t_i and a is the initial error rate at time t = 0. The function Z_m is not the same function as λ_m, but in the case of the exponential model, Z_m can be transferred to λ_m.

STATIC ESTIMATION

Least Square Method

The static least square method, which is used as one of the apriori estimates, is based upon Taylor series expansion of learning at points of estimation. Using the static exponential model for human hazard rate and equating to zero the partial derivatives of error squared with respect to each of the model parameters, an updated estimate can be obtained. This process is repeated until the convergence is achieved.

Impulse Moment Updating

Impulse moment updating is based on the estimation of the system transfer function parameters by output; input division in "S" domain or output-input synthetic division in time domain. This method was first introduced by Ba Hli (7, 8). Using the exponential hazard function and considering a unit step as input, the following result can be obtained:

$$y_f = \sum_{i=1}^{N} \{h\}_{A_1} \tag{8}$$

$$\tau = \frac{\sum_{i=1}^{N} \bar{t}_i \{h\}_{A_i}}{\sum_{i=1}^{N} \{h\}_{A_i}} \tag{9}$$

$$\bar{t}_i = \frac{t_i + t_i+1}{2} \qquad (10)$$

$$\{h\}_A = \frac{\{^f0\}}{\{f_i\}} \qquad \text{synthetic division} \qquad (11)$$

where y_f is the observation at (t=∞) minus the observation at (t=0) and τ is the time constant of the exponential.

KALMAN FILTERING (DYNAMIC ESTIMATION)

To use Kalman filtering for dynamic estimation, it is necessary to construct a dynamic model for a hazard function. [9] This can be done by a variety of methods [2, 10]. For a hazard function defined as in Eq. (4) the following dynamic model can be constructed;

$$\begin{bmatrix} \lambda \\ \alpha \\ c \end{bmatrix}_{t+\Delta t} = \begin{bmatrix} 1+\hat{\alpha}\Delta t & \hat{\lambda}\Delta t & \Delta t \\ 0 & 1 & 0 \\ 0 & 0 & 1 \end{bmatrix} \begin{bmatrix} \lambda \\ \alpha \\ c \end{bmatrix}_t + \begin{bmatrix} \hat{\alpha}\ \hat{\lambda}\ \Delta t \\ 0 \\ 0 \end{bmatrix}_t \qquad (12)$$

$$Z_t = [1\ 0\ 0] \begin{bmatrix} \lambda \\ \alpha \\ c \end{bmatrix} \qquad (13)$$

where;

$$c = a/\tau \qquad (14)$$

This system is neither observable nor controllable, so the application of Kalman filtering for estimation of λ, α, and c should be done carefully to prevent filter instability. A complete stability analysis was done to determine the initial values for the covariance matrices. Also, the concept of forward, backward and smoothed filtering were applied.

RESULTS AND CONCLUSIONS

Table 1 and Figure 1 show estimations for the human hazard function parameters and the transformed estimates of hazard functions for PWR's using different static estimator as initial feed to the Kalman filter program. It can be seen that Kalman filtering estimation is not sensitive to different initial values.

In both cases of PWR's and BWR's, the learning effect is observed by the characteristic of the hazard function for three different facility sizes. It is concluded that for PWR's there is a direct correlation between operator error rate and facility size; the larger the PWR, the greater the number of errors committed. While for BWR's reactor size does not seem to correlate with operator error rates.

The Kalman filtering computer module developed here is found feasible for error data updating. In fact, a Kalman dynamic system may be developed, stored, and continuously updated to monitor error rates, whether in a specific plant or in the whole industry [1]. The input to this system can be directly obtained by LERRET, which has been developed to retrieve LER data [1]. The hazard function model tested here can be used to estimate operator learning rates. However, using constant parameters for the exponential performance functions provides a poor estimation.

Table 1. *Dynamic estimation for PWR's*

Initial static model	impulse moment updating			Least square method		
Time	a	b	$1/\tau$	a	b	$1/\tau$
0	.02118	23.01	0.05092	.02314	7.498	0.06964
4	.02118	24.46	0.05092	.02314	16.73	0.08645
8	.02118	20.00	0.08058	.02314	17.17	0.07919
12	.02118	22.14	0.07325	.02314	19.38	0.07224
16	.02118	21.80	0.07202	.02314	19.00	0.07076
20	.02118	22.82	0.07013	.02314	19.90	0.06880
24	.02118	24.21	0.06852	.02314	21.13	0.06711
28	.02118	25.21	0.06733	.02314	21.99	0.06584
32	.02118	26.43	0.0659	.02314	23.04	0.06429
36	.02118	27.28	0.06449	.02314	23.76	0.06277
40	.02118	27.93	0.06307	.02314	24.29	0.06125
44	.02118	28.13	0.0622	.02314	24.41	0.06035
48	.02118	27.95	0.06207	.02314	24.22	0.06024
52	.02118	27.83	0.06205	.02314	24.08	0.06025
56	.02118	27.87	0.06194	.02314	24.09	0.06016
60	.02118	27.83	0.06195	.02314	24.03	0.06019
64	.02118	27.65	0.06207	.02314	23.84	0.06033
68	.02118	27.56	0.06211	.02314	23.73	0.06039
72	.02118	27.37	0.06219	.02314	23.53	0.06047
76	.02118	27.22	0.06223	.02314	23.37	0.06051
80	.02118	27.49	0.06214	.02314	23.6	0.06043
84	.02118	27.38	0.06217	.02314	23.48	0.06046
88	.02118	27.20	0.06221	.02314	23.28	0.06051
92	.02118	26.86	0.06227	.02314	22.91	0.06058
96	.02118	26.86	0.06227	.02314	22.91	0.06058
100	.02118	26.81	0.06228	.02314	22.84	0.06059
104	.02118	27.19	0.06221	.02314	23.17	0.06052
128	.02118	26.85	0.06226	.02314	22.84	0.06059
152	.02118	27.26	0.06226	.02314	23.16	0.06059
156	.02118	27.27	0.06226	.02314	23.16	0.06059

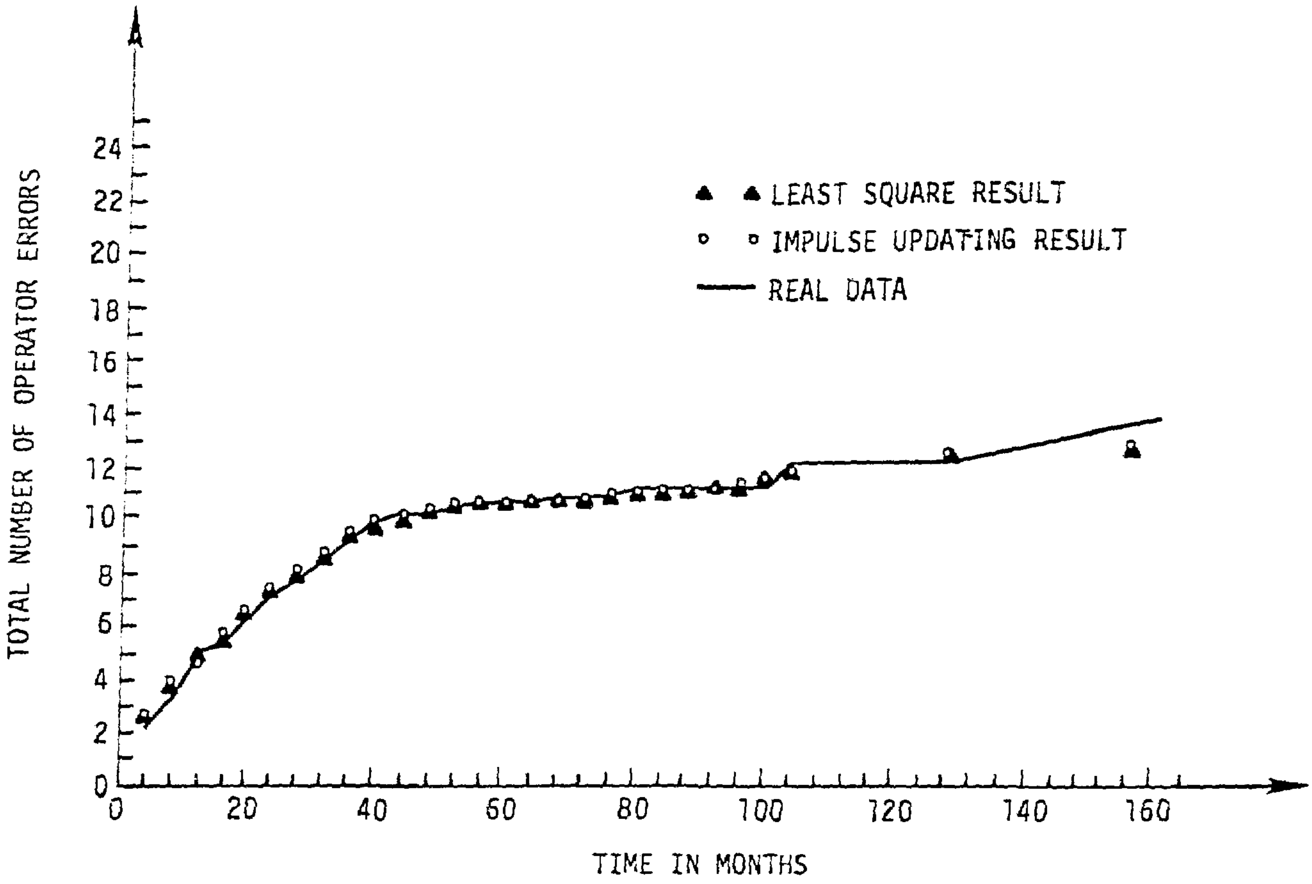

Figure 1. Dynamic estimation for average PWR's power plant.

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IDENTIFICATION OF PROGRAMS FOR IMPROVED OPERATOR PERFORMANCE BASED ON PAST EXPERIENCES IN COMMERCIAL LIGHT WATER REACTORS

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ABSTRACT

Over 11,000 Licensee Event Reports (LER's) were manually reviewed for all Light Water Reactors (LWR's) from the period January 1972 through December 1977. Five hundred and twelve total operator errors (OE's) were identified, accounting for approximately 3.0 and 2.6 OE's per PWR and BWR reactor year of operation, respectively. Specific failure modes resulting in OE's have been identified with respect to systems and components involved. Operator error significance has been qualitatively determined. The development of frequency/consequence matrices has enabled an importance ranking for operator improvement programs.

INTRODUCTION

Recently much concern has focused on the effects of the human operator on nuclear electric power plant (NEPP) operations. Both the operator and the human-system interface play a major role in the safety, reliability, and availability of NEPP systems. The results of the reactor safety study reported in WASH-1400 [1] emphasized the need for actuarial data on operator reliability for analysis of the safety impact of the operator on the nuclear systems.

Eleven thousand, two-hundred and seventy-six Licensee Event Reports (LER's) from January 1972 through December 1977, for all U.S. commercially operating NEPP's, have been manually reviewed to identify frequently occurring events initiated by the operator. Thirty-seven Pressurized Water Reactors (PWR's) and 26 Boiling Water Reactors (BWR's) accumulated 102.63 and 80.33 reactor-years of operation, respectively, based upon yearly plant availability factors during the data acquisition period.

Five hundred and twelve operator errors (OE's) were identified which account for 3.02 OE's per PWR reactor-year of operation and 2.63 OE's per BWR reactor-year of operation. These errors do not include human errors in the stages of design, fabrication, maintenance and testing, or installation. The operators considered here are senior control operators, control operators, and equipment operators; including the radwaste system operators.

OPERATOR ERROR FREQUENCY

Table 1 categorizes operator error frequency according to general systems affected, while Table 2 delineates system specific components involved in OE's. It is apparent, for example, that valves in Emergency Core Cooling Injection systems (ECCIS) are more frequently involved in OE's in both PWR's and BWR's, and improper interactions with switches, pumps, breakers, and tanks make up the majority of the balance of OE's in the ECCIS.

These numbers do not specifically address the severity or the consequences of operator errors, but do include significant, potentially significant, and insignificant errors. Table 3 lists 10 components involved in 75% of the total operator error population, and the most frequent operator error--failure modes (FM). In general, it is seen that frequent failure modes for components which require direct interaction (e.g., valves, switches, breakers) result from leaving a component in the wrong position, component mispositioning, or component deenergization, while indirect component interaction (e.g., monitors, control rods, pumps, etc.) failure modes are usually due to not checking/testing component operability, or exceeding technical specification functional limits.

SIGNIFICANCE OF OPERATOR ERRORS

Operator error frequency can be put into perspective by categorizing the significance of error occurrences. Although the evaluation of significance is somewhat subjective and rather difficult to determine from LER's alone, it is a necessary step for the identification of the impact of operator errors on the safe operation of NEPP's. The significance classification scheme used here has been largely based on definitions in WASH-1314, [2] and system descriptions available in WASH-1400. Three degrees of significance were identified and included:

1. Directly significant

   - Significant property damage or personal injury
   - Release of radioactivity from the site in excess of technical specifications (tech specs)
   - Violation of safety limits set forth in tech specs
   - Loss of significant engineered safeguard systems while operating

2. Potentially significant

   - Uncontrolled or unplanned release of radioactive material from the site in amounts less than those allowed by technical specifications
   - Equipment malfunctions in primary coolant systems or engineered safety features
   - Errors resulting in shutdowns, delays, or adverse plant performance
   - Unusual conditions such as transients for which response was different than expected
   - Violation of limiting conditions of operation or limiting safety systems setpoints

Table 1. Frequency of Operator Errors with Respect to System Involved.

System	Errors per Reactor Year of Operation		
	BWR	PWR	Total LWR
Emergency Core Cooling Injection Systems (ECCIS)	0.46	0.72	1.18
Waste Processing Systems (WPS)	0.42	0.34	0.76
Instrumentation and Monitoring Systems (IMS)	0.42	0.26	0.68
Reactor Power Control Systems (RPCS)	0.29	0.31	0.60
Secondary Non-nuclear Systems (SNS)	0.12	0.29	0.41
Reactor Protection Systems (RPS)	0.25	0.16	0.41
Auxiliary Electric Power Systems (AEPS)	0.19	0.19	0.38
Main Reactor Cooling Systems (MRCS)	0.12	0.19	0.31
Auxiliary Systems for Normal Operation (ASNO)	0.05	0.22	0.27
Other Containment Systems (OCS)	0.15	0.13	0.28
Containment Spray and Recirculation Systems (CSRS)	0.01	0.20	0.21
Emergency Cooling Recirculation Systems (ECRS)	0.14	0.0	0.14
TOTAL	2.62	3.01	5.63

Table 2. Frequency of Operator Error with Respect to Specific Components Involved*.

	SYSTEM																							
COMPONENT	AEPS		RPS		ECCIS		ECRS		CSRS		OCS		MRCS		RPCS		ASNO		SNS		WPS		IMS	
	B$^+$	P^{++}	B	P	B	P	B	P	B	P	B	P	B	P	B	P	B	P	B	P	B	P	B	P
Valve	.01	-	.06	-	.22	.29	.05	-	-	.12	.06	.09	.02	.08	-	.01	-	.14	.05	.10	.24	.19	.1	-
Monitor/Detector	-	-	-	-	-	.01	.04	-	-	-	-	-	-	-	-	-	-	.02	-	.02	.02	.03	.20	.13
CR/CR Drive	-	-	.12	.01	-	-	-	-	-	-	-	-	-	-	.16	.16	-	-	-	-	-	-	-	-
Switch	.01	.01	.01	.03	.06	.03	-	-	.01	.01	.01	.08	.01	-	.01	.01	.02	.01	.04	.01	.01	-	.04	.04
Pump	-	-	-	-	.03	.10	-	-	-	.02	-	-	.05	.02	.05	-	-	-	.01	.06	.01	.01	.01	-
Breaker	.01	.04	-	.01	.05	.04	-	-	-	.05	-	.01	-	-	-	.01	-	-	-	.02	-	-	-	-
Tank	-	-	-	-	.01	.15	-	-	-	-	-	-	-	-	-	-	-	-	-	-	.04	.05	-	-
Alarm	-	.01	-	.02	-	-	-	-	-	-	.01	-	-	-	-	-	-	-	-	-	.02	-	.01	.03
Diesel Generator	.05	.04	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	.01	-	-	-	-
Bus	.04	.05	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Jumper	-	-	-	-	.01	-	-	-	-	-	-	.01	-	-	-	-	-	-	-	.01	-	-	.01	-
Transmitter	-	-	-	.04	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Recorder	-	-	-	-	-	-	.01	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	.02	.01
Relay	-	.01	-	-	.01	.01	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Bistable	-	-	-	.02	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
TIP	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	.01
Battery	.02	.01	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	.02	.01

- Implies that no operator error data was recorded for system/component combination

\+ BWR error rate per reactor year of operation

++ PWR error rate per reactor year of operation

Table 3. Frequent Operator Failure Modes.

Component Involved	Failure Mode	Number of Events			% of Total Component Occurrences[a]			% of Total Operator Errors[b]		
		BWR	PWR	LWR	BWR	PWR	LWR	BWR	PWR	LWR
Valve	Left in wrong position	24	43	67	41	41	41	11	14	13
	Misalignment	17	16	33	29	15	20	8	5	6
	Inadvertant actuation	7	9	16	12	9	10	3	3	3
	Improper operation	5	7	12	8	7	7	2	2	2
	Procedure	2	5	7	3	5	4	1	2	1
Monitor/ Detector	Did not check/test	8	5	13	38	24	31	4	2	2
	Did not monitor	5	5	10	24	24	24	2	2	2
	Left in/out of service	3	2	5	14	10	12	1	1	1
Control Rod/Control Rod Drive/ Control Rod Group	Did not check/test	7	1	8	30	6	20	3	0	2
	Procedure	6	2	8	26	12	20	3	1	2
	Exceeds limits	0	6	6	0	35	15	0	2	1
	Improper sequence	2	1	3	9	6	8	1	0	1
Pump	Improper operation	4	8	12	31	38	35	2	3	2
	Did not start/stop	3	2	5	23	10	15	1	1	1
	Procedure	2	2	4	15	10	12	1	1	1
	Did not check/test	1	3	4	8	14	12	0	1	1
Switch	Left in wrong position	3	8	11	15	47	30	1	3	2
	Mispositioning	7	2	9	35	12	24	3	1	2
	Inadvertant actuation	5	1	6	25	6	16	2	0	1
Breaker	Inadvertant actuation	2	5	7	33	26	28	1	2	1
	Deenergization	1	5	6	17	26	24	0	2	1
	Did not check/test	2	2	4	33	11	16	1	1	1
Tank	Exceeds limits	2	6	8	50	30	33	1	2	2
	Did not monitor	1	3	4	25	15	17	0	1	1
	Misinterpretation	1	2	3	25	10	13	0	1	1
Alarm	No response	3	4	7	75	67	70	1	1	1
Diesel Generator	Did not check/test	1	3	4	25	60	44	0	1	1
	Disengaged	1	1	2	25	20	22	0	0	0
Bus	Deenergization	2	2	4	67	40	50	1	1	1
	Did not check/test	0	2	2	0	40	25	0	1	0

[a] $\frac{\text{No. of specific component/failure mode events}}{\text{Total No. of specific component events/reactor type}}$ X 100% : To nearest whole percent

[b] $\frac{\text{No. of specific component/failure mode events}}{\text{Total No. of operator errors/reactor type}}$ X 100% : To nearest whole percent

3. Insignificant

   - Clearly no adverse real or potential effect on plant safety but which constitutes a literal violation of the technical specifications

FREQUENCY/CONSEQUENCE MATRIX

Categorization of operator errors involved with the ECCIS have been made delineating reactor type, component, failure mode and significance for the most frequently occurring events previously identified. The results are shown in the Frequency (F), Consequence (C) matrices of Figures 1 and 2 for BWR's and PWR's, respectively. Consequence has been divided according to signfica e, as previously suggested. Frequency ranges are based upon the following:

- High Frequency
  -One specific event occurs in one year of operation among the 26 BWRs in the data base
- Medium Frequency
  -One specific event occurs in two years of operation among the 26 BWRs in the data base
- Low Frequency
  -One specific event occurs in four years of operation among the 26 BWRs in the data base.

PWR frequency delineation was made on the same basis of 26 plants, for comparative purposes.

Table 2 shows that valving errors in BWR's are about seven times more frequent than operator errors involving pumps, while Figure 1 indentifies the two types of events as having equivalent significant importance (R_s) when importance, R, is defined as

$$R = F \times C . \qquad (1)$$

Figure 1 illustrates that

$$R(V)_s = R(P)_s = R(B)_s = R(W)_s \qquad (2)$$

where

V = Valve; P = Pump; B = Breaker; W = Switch;

s = Significant Matrix Column

and

$$R(V)_p \simeq 5\ R(B)_p \simeq 5\ R(W)_p \simeq 9\ R(P)_p \qquad (3)$$

Frequency (F_B) (Operator errors/BWR reactor year of operation)	Insignificant (N)	Potentially Significant (p)	Significant (s)
≤.01			
$.01<F_B<.04$	Valve: Inadvertent actuation (.012) Overtorqued (.012) Misalignment (.012) Tank: High/low level (.012) Breaker: Did not test (.012) Switch: Left in wrong position (.012) Inadvertent actuation (.012)	Valve: Inadvertent actuation (.025) Overtorqued (.012) Pump: Did not start/stop (.012) Breaker: Inadvertent deenergization (.012) Left deenergized (.012) Switch: Left in wrong position (.012) Inadvertent actuation (.012)	Valve: Left in wrong position (.012) Pump: Improper flow rate (.012) Breaker: Inadvertent deenergization (.012) Switch: Improper set points (.012)
≥.04	Valve: Left in wrong position (.05)	Valve: Left in wrong position (.075)	

Consequence (C)

Figure 1. Frequency/Consequence Matrix for Operator Errors in BWR Emergency Core Cooling Injection Systems

Frequency (F_p) (Operator Errors/PWR Reactor Year of Operation)	Insignificant (N)	Potentially Significant (p)	Significant (s)
$\leq .01$	Valve: Incorrect actuation (.010) Pump: Not returned to service (.010) Did not test/check (.010) Breaker: Failure to lock open/closed (.010) Switch: Incorrect operation (.010)	Valve: Negligence (.010) Incorrect actuation (.010) Inadvertent actuation (.010) Overtorqued (.010) Pump: Improper flow rate (.010) Not returned to service (.010) Breaker: Improper set points (.010) Switch: Incorrect operation (.010) Inadvertent actuation (.010)	Valve: Inadvertent actuation (.010) Tank: Inoperable (.010) Pump: Not returned to service (.010)
$.01<F_B<.04$	Valve: Misalignment (.029) Left in wrong position (.029) Tank: Improper fill rate (.019)	Valve: Misalignment (.019) Pump: Incorrect operation (.029) Did not start/stop (.019) Breaker: Inadvertent deenergization (.019)	Valve: Left in wrong position (.019)
$\geq .04$	Tank: High/low level (.058) Improper concentration (.058)	Valve: Left in wrong position (.136)	

Consequence (C)

Figure 2. Frequency/Consequence Matrix for Operator Errors in PWR Emergency Core Cooling Injection Systems

where

p = Potentially Significant Matrix Column .

Results as given (from Figure 1 and Equations 2 and 3) identify the relative importance (R) of component/failure mode combinations involved in operator error.

RANKING OF IMPROVEMENT PROGRAMS

A Frequency/Consequence matrix methodology has been utilized (based upon the Value/Impact approach) to identify operator performance improvement programs that should have implementation priorities. If we assume that potentially significant (p) and significant (s) events are of primary concern (i.e., $R_i << R_s + R_p$) and that some quantitative measure of "s" is greater than "p" (and defineable), then a total measure of importance can be placed upon components to determine specific programs for improved operator performance. For example, R(V) and R(B) can be defined as

$$R(V) = R(V)_s + R(V)_p = .012s + (.075 + .025 + .012)\,p \qquad (4)$$

$$R(B) = R(B)_s + R(B)_p = .012s + (.012 + .012)p \ . \qquad (5)$$

Their ratio R(V)/R(B) is then an overall measure of relative amounts of some quantity (e.g., time, money) which should be appropriately invested to improve operator performance, and the F/C matrix delineates the specific operator actions which should be improved. As an example, if s = 4p, then from Eqs. (4) and (5) for BWRs:

$$\frac{R(V)}{R(B)} = \frac{.012s + (.112) \times .25s}{.012s + (.024) \times .25s} = \frac{(.012 + .0112)s}{(.012 + .0024)s} \simeq 2$$

which suggests that twice as much time, money, etc. should be invested into programs to decrease OE's involved with

- Leaving valves in wrong position;
- Inadvertent valve actuations; and
- Overtorquing of valves;

as is invested into programs to decrease OE's involved with

- Inadvertent deenergizations of breakers; and
- Leaving breakers deenergized.

It is apparent that the ratio s/p is subjective, but even if an s/p range is utilized, say 2-10, a rank ordering of operator improvement programs is possible. Table 4 lists the suggested order of implementation of programs to increase operator performance in the ECCIS.

Table 4. Suggested Order of Operator Improvement Program Topics for the ECCIS

Implementation Order	Involved Component	Failure Mode
BWR		
1	Valve	Left in wrong position Inadvertent actuation Overtorqued
2	Breaker	Inadvertent deenergization Left deenergized
2	Switch	Improper set points Left in wrong position Inadvertent actuation
3	Pump	Improper flow rate Did not start/stop
PWR		
1	Valve	Left in wrong position Inadvertent actuation Misalignment Incorrect actuation
2	Pump	Not returned to service Incorrect operation Did not start/stop Improper flow rate
3	Tank	Inoperable
4	Breaker	Inadvertent deenergization Improper set points
5	Switch	Incorrect operation Inadvertent actuation

CONCLUSION

Analysis of operator errors in LWR's has shown that 24% and 18% of all OE's are involved with Emergency Core Cooling Injection Systems for PWR's and BWR's, respectively, and that other systems more frequently involved include Waste Processing, Instrumentation, and Control, and Reactor Power Control Systems.

Components most frequently involved in OE's include valves (22% of OE's), control rods (6%), monitors (5%), pumps (5%) and switches (5%).

Calculated OE error rates, with respect to error significance, has enabled the development of Frequency/Consequence matrices. In this manner exact quantification of Frequency is not necessary for matrix utilization.

The development of Importance Factors has identified a rank ordering for operator performance improvement programs, with valve mismanipulations being of first priority. Results generally show that acts of omission are of principal concern with valves and that acts of commission require training program consideration for breakers, switches, and pumps.

Additional efforts will quantify the relationship among significance factors.

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2. Thompson, D., E. L. Boyle, and L. H. Grosman, 1974. Summary of Abnormal Occurrences Reported to the AEC during 1973, WASH-1314, USAEC.

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RESPONSE TREES FOR EMERGENCY OPERATOR ACTION AT THE LOFT FACILITY

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ABSTRACT

A technique for assisting nuclear plant operators during emergency conditions has been developed and implemented at the LOFT facility. The technique is based on "response trees". A response tree is a diagram showing the modes available for responding to an accident and the relative desirability of each. A procedure using response trees is a central reference which directs the operator to specific procedures for responding to the accident. Benefits of the technique include 1) it facilitates efficient operator response, 2) it encourages operator familiarity with all accident response modes, and 3) it applies to many accidents, including common mode and multiple failure events.

INTRODUCTION

Following the onset of an accident which disables equipment used for normal reactor cooling, the first priority of the nuclear plant operator is to ensure that the reactor core is covered with water and that adequate cooling water flow is established. During this time, he must evaluate the situation, determine which emergency procedures apply, find the appropriate procedures, and perform the prescribed actions. Failure to respond quickly and effectively could result in expensive facility damages and potential hazards to the public. A procedure which attempts to streamline this short-term response process has been developed and implemented for the Loss-of-Fluid Test (LOFT) facility.

RESPONSE TREES FOR LOFT

The procedure developed for LOFT is entitled "Loss of Normal Decay Heat Removal Modes." Diagrams called "response trees" have been included in the procedure to illustrate potential modes for cooling the reactor and the relative priority for using each. The procedure is designed to be a central reference point to be used by the operator to determine which specific emergency procedures should be used to respond to the accident.

Figure 1 is the response tree for the LOFT Low Pressure Injection System (LPIS), and Figure 2 is a simplified schematic of the LPIS. The response tree shows all potential cooling modes available using the Low Pressure Injection System. Each cooling mode has five elements: a heat sink, a water source, a pump, a route, and an injection point. Each element may represent many individual components.

The five elements are shown on the various levels of the response tree. Each path from the bottom of the tree to the top represents a different cooling mode. At the bottom of each path (cooling mode) is listed a priority number and a reference to the appropriate procedure(s) in the LOFT Plant Operating Manual (POM). Priority numbers were established by evaluating the relative desirability of the cooling modes in terms of cooling effectiveness, difficulty of implementation, and other similar considerations. Cooling modes with small priority numbers are most desirable, and cooling modes which may be initiated automatically are so labeled. At appropriate points on each cooling mode are listed pressure (PI) and flow (FI) instruments which can be used to monitor the performance of the cooling mode.

USE OF THE PROCEDURE

Following the onset of the accident, the operator immediately refers to the procedure to determine an appropriate course of action. Using his current knowledge of system status, he crosses out or otherwise indicates any components which he knows to be disabled. He then does the same for all priority numbers of cooling modes which require the use of a disabled component. Next, he selects from the remaining cooling modes the one(s) with the smallest priority number, refers to the listed procedure(s), and performs the prescribed actions. For example, if LPIS pump A fails to start, a pressure indicator in the downcomer injection line indicates that flow is not reaching the reactor vessel, and the Borated Water Storage Tank (BWST) is empty, he selects the cooling mode with priority number 6, refers to POM procedure 9.4.10, and performs the appropriate actions (see Figure 3). As time progresses and other components are disabled or restored, he continually updates the response tree to ensure that the optimum cooling mode is being implemented.

COLOR GRAPHICS DISPLAY

A color cathode ray tube (CRT) display is being developed for this procedure in conjunction with the LOFT Augmented Operator Capability Program. Figure 4 shows the display as it will look for the example accident. Unavailable components will be shown in magenta, available components will be shown in dark blue, and the recommended cooling mode will be highlighted with double-width lines in cyan. A computer will be used to monitor system status, evaluate the response tree, and generate the correct CRT display for the recommended response.

ADVANTAGES OF THE TECHNIQUE

The following strengths have been noted in the development and implementation of this technique at the LOFT facility:

- It provides a systematic method for identifying all potential cooling modes, establishing their relative priority, and displaying this information for operations personnel.

- Rather than requiring the operator to refer to the entire POM for an applicable procedure, it provides a central point from which he is referred directly to the correct procedure.
- It improves operator familiarity with all potential modes for cooling the reactor and the interrelationships between plant systems and components.
- It is relatively simple and inexpensive to implement.
- The trees are easily modified if facility modifications occur.

CONCLUSION

The use of this technique for accident response can provide the immediate actions necessary to bring the system under control. Sophisticated fault-isolation techniques could then be used to determine the exact cause of the accident and optimize the ultimate recovery of the facility. Thus, response trees could prove to be an important element in responding effectively to nuclear reactor accidents.

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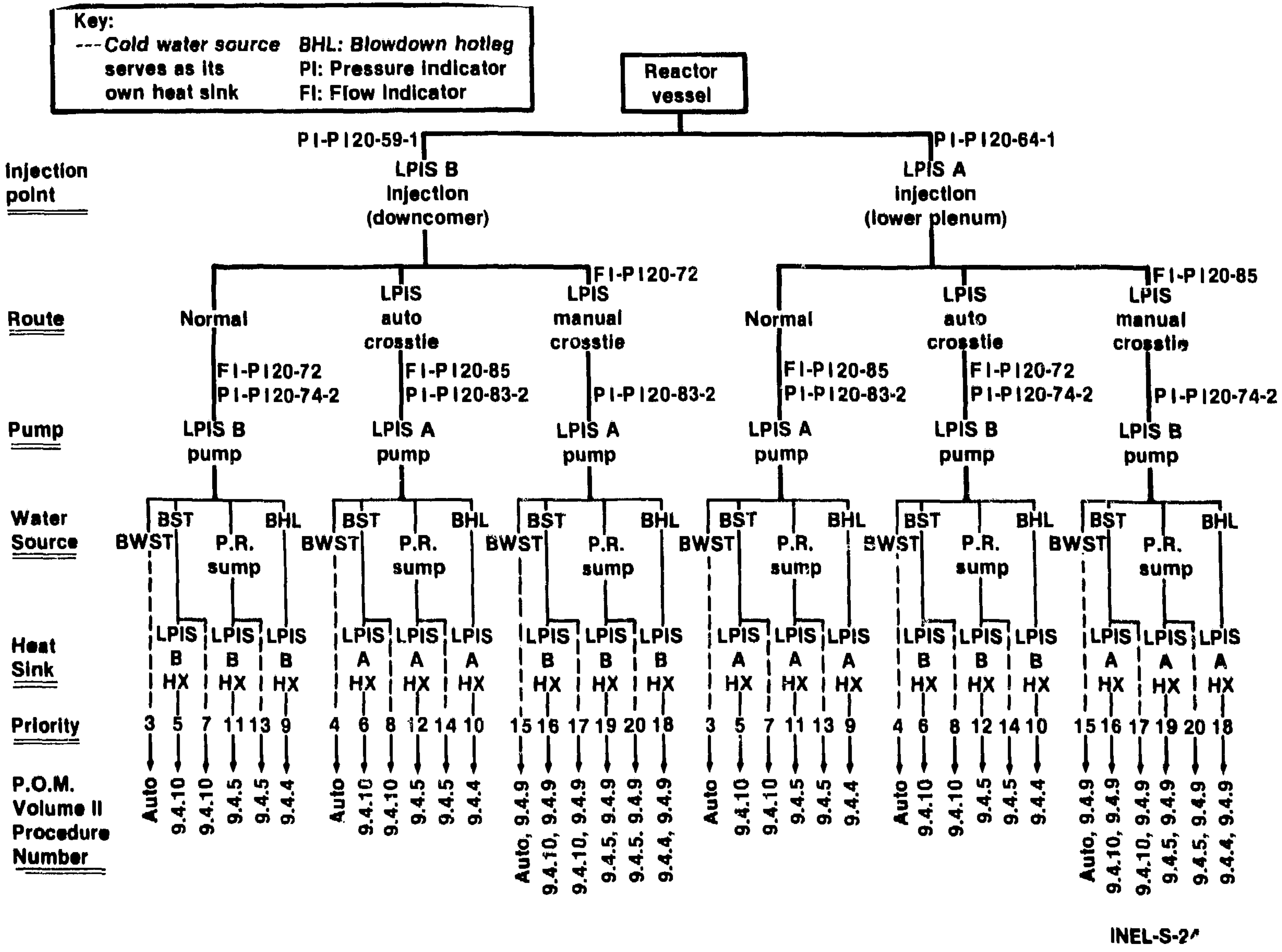

Figure 1. LOFT LPIS Response Tree.

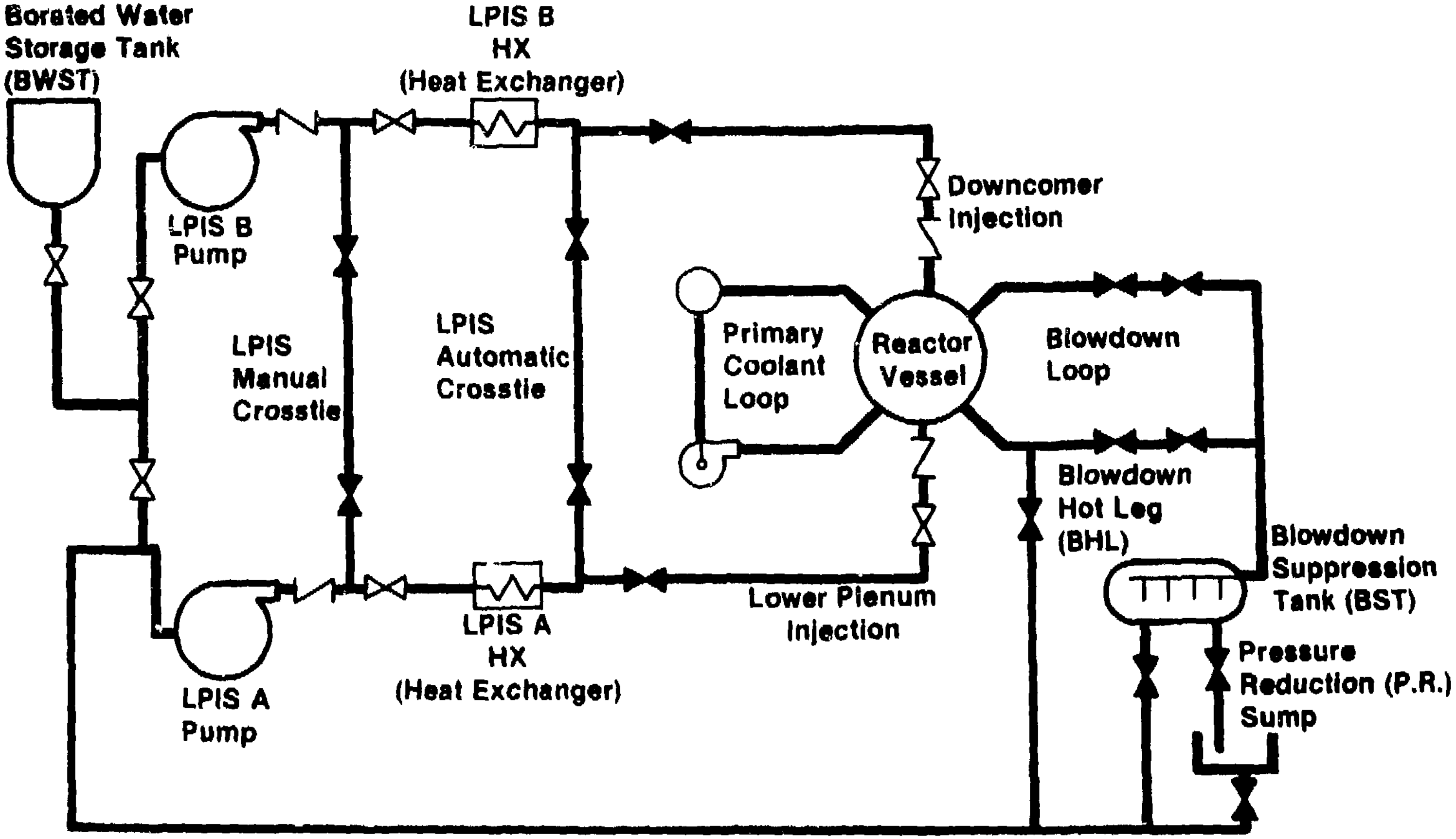

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Figure 2. Simplified Schematic of LOFT Low Pressure Injection System (LPIS).

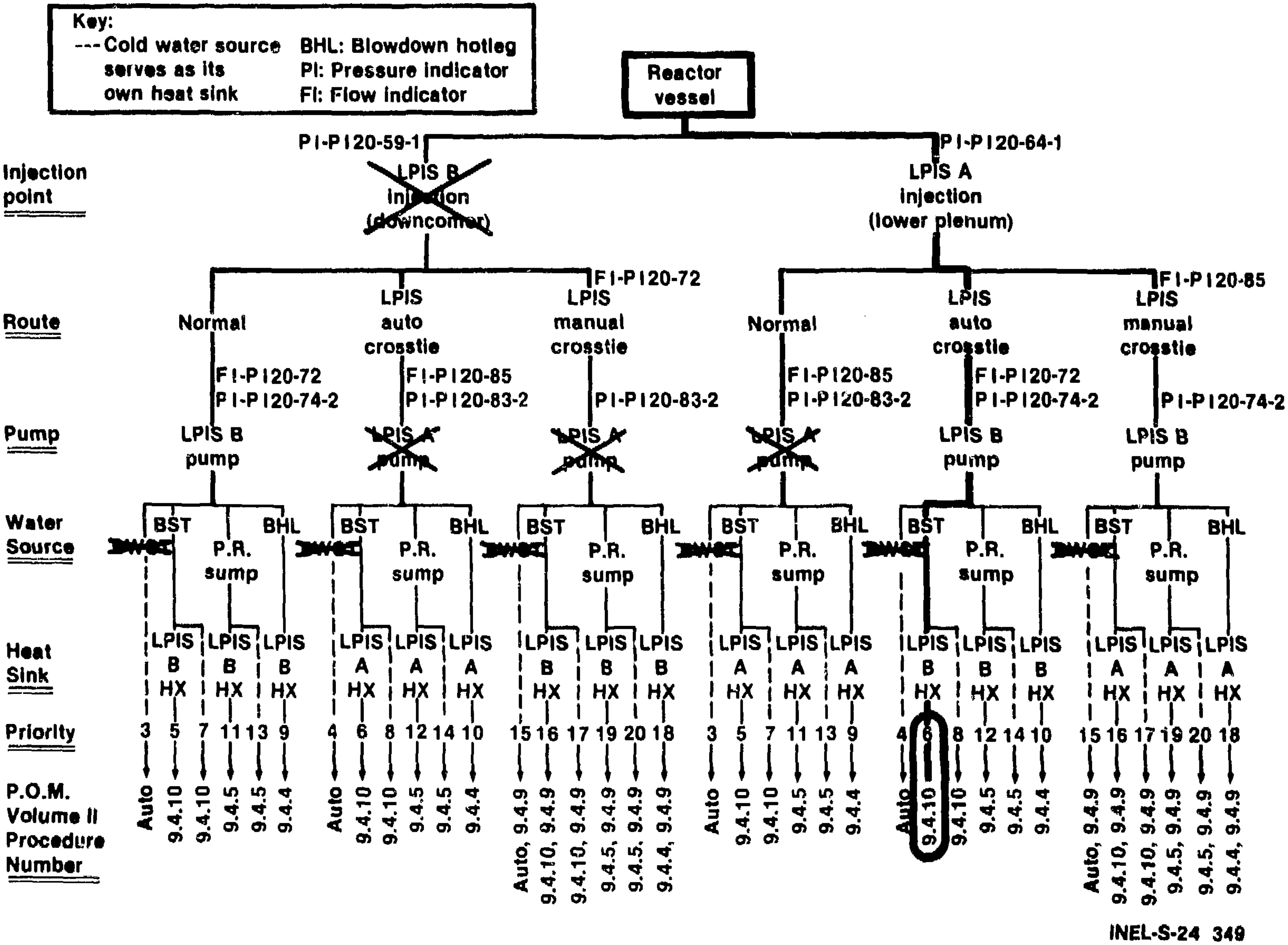

Figure 3. Choice of Cooling Mode for Example Accident.

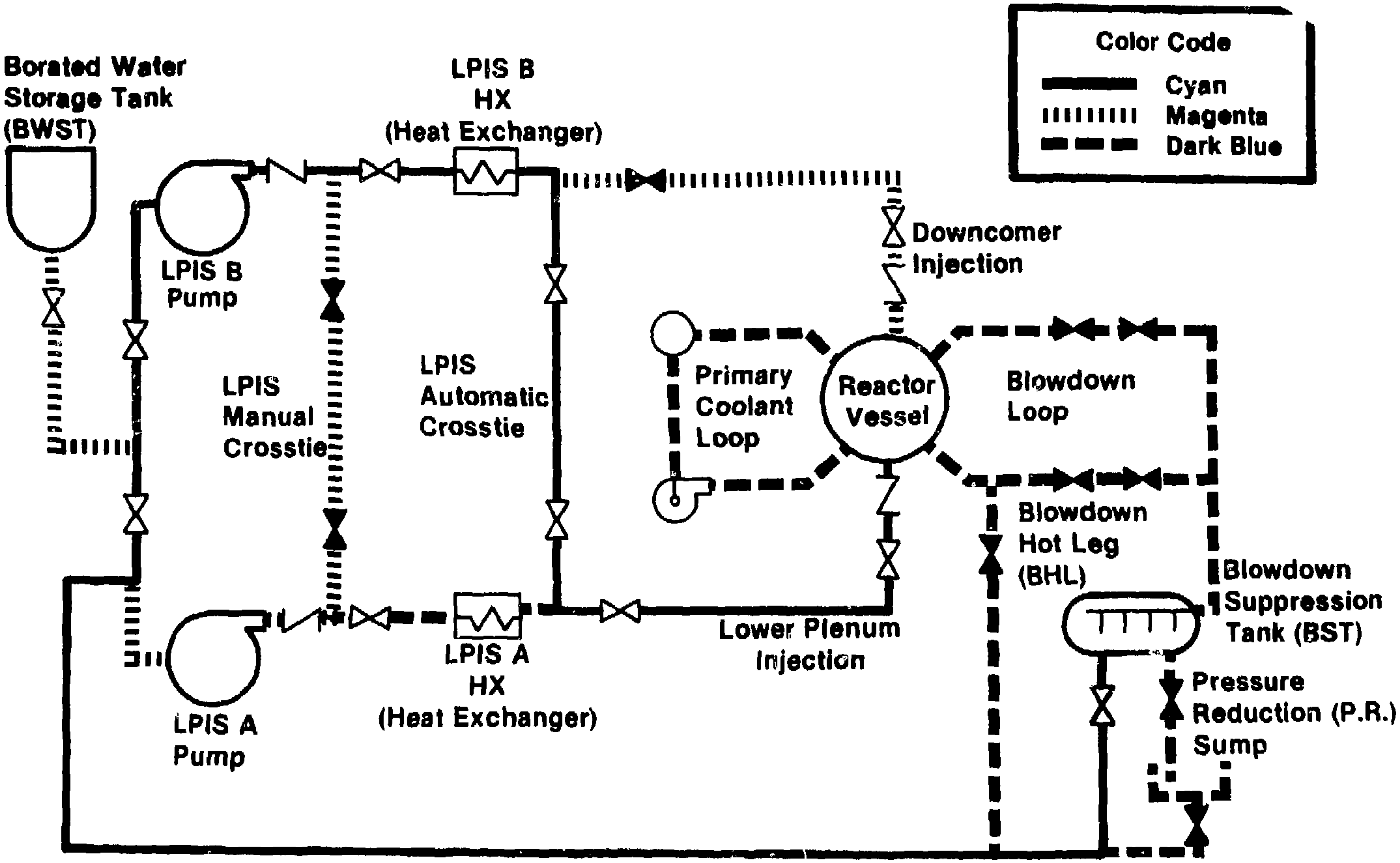

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Figure 4. Color CRT Display for Example Accident.

A METHOD FOR ANALYSING INCIDENTS DUE TO HUMAN ERRORS ON NUCLEAR INSTALLATIONS

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ABSTRACT

This report deals with the development of a methodology adapted to a detailed analysis of incidents considered as due to human errors.

The method is illustrated by the analysis of two reactor incidents, one during the handling of a heat exchanger, the other during the refueling of a reactor. In studying the first incident, we show how to build an event sequence, a fault tree and a simplified fault tree in order to highlight the elements which are situated at the origin of the incident.

In studying the second incident, we show how to analyse a human failure, in order to decompose the mechanisms and the causes of this failure.

INTRODUCTION

An incident is the result of inappropriate actions by either components or human elements of the installation. These actions are respectively called human failures and component failures.

We can identify different human failures according to the various tasks to be performed by man. Some may have a direct influence on the behaviour of the plant(for example, human failure in executive operation), some may have an indirect influence (human failure in transmission of information) and others a combined influence (human failure in control task).

After identifying the human failures, we have to analyse the characteristics of the work which have caused each of them : for example, the work organization, the design of the work station, the training of the personnel, etc... More details on these different causes are given in reference [1].

All these principles are presented on fig. 1.

From this figure, two main analysis steps appear :
- identification of the failures
- determination of their causes related to human elements.

These two steps are described in the following paragraphs and are illustrated by two reactor incidents.

1. IDENTIFICATION OF THE FAILURES

This first step is decomposed into two phases :

- determination of the events which have generated the incident, leading to the construction of an event sequence,
- identification of the events which are specific failures, leading to the construction of a fault tree.

1.1. Construction of an event sequence

In constructing the event sequence, we take into account three types of events : the state of the components and their relative actions, and the actions performed by the human elements. We stop the decomposition of the event sequence at the levels of permanent state or usual action.

In order to illustrate these principles, we give the example of an incident occured during the handling of a heat exchanger. This incident is simplified for greater clarity of the paper : for more detailed information, one should read [2].

This incident corresponds to the deformation of a heat exchanger while it was introduced into the reactor. The event sequence is presented on fig. 2.1. : the incident is due to a human action when the operator chose the high speed in downward motion, combined with a component action when the heat exchanger grazed into the reactor. The first event is a usual action, so we stop the genesis at this level. The other one is an unusual action, which is due to a human action in moving the exchanger, combined with a component state "small clearance between exchanger and reactor" and another component state "exchangerout of axis". The first state is a permanent one, so we stop the genesis at this level. In order to simplify this illustration of an incident analysis, we don't continue the construction of the genesis of other events.

1.2. Construction of a fault tree

After the construction of the event sequence, we can identify the events which are specific failures. As a failure is defined by a discrepancy between the performed action and the desirable action which would have avoided the incident, we will only take into account the events which are actions. So, on fig. 2.2., we have two human failures and one technical failure. In transforming the whole sequence into a fault tree, the perturbed state (exchangerout of axis) becomes a "and" gate, and the permanent state (small clearance between exchanger and reactor) belongs to the "technical design" of the plant.

In simplifying the failure tree, we can cut out those failures entirely explained by previous elements. So, the technical failure is explained by the human failure (in moving the exchanger), the "technical design" and the "and" gate. For each human failure, we have to define if there is a discrepancy between the written procedure and the performed action. If there is no discrepancy, we can entirely explain the human failure by the "work" organization ; this is so for the human failure when choosing the high speed in downward motion.

The simplified tree presented on fig. 2.3. highlights the elements which are situated at the origin itself of the incident:"work organization", "technical design", human failure and a "and" gate which would be explained by constructing the whole genesis of the incident.

2. DETERMINATION OF THE CAUSES OF HUMAN FAILURES

The analysis of a human failure consists in an explanation of the discrepancy between the accomplissed task and the desired task. This analysis can be decomposed into the following phases :

- determination of the task concerned by the failure : is it an executive task, a control task, etc... ?
- determination of the operation concerned by the failure : is it during taking information,diagnosis, decision or action ?
- identification of the mechanisms of the failure : is it an omission, a confusion, etc... ?
- identification of the causes of the failure : is it due to the work organization, the social environment, the work station design, etc... ?

In order to illustrate these principles, we give the example of a human failure occured during the refueling of a reactor. This failure is simplified for greater clarity of the paper : for more detailed information, one should read [3] .

The human failure consisted in an error in addressing one element of the reactor core. The refueling shift consisted of two operators : the first one had to take information and to transmit this information, the second one had to control the information and to execute the refueling task. After many interviews, we found that there was a combination of a failure in transmitting information (operator 1) and a failure in controlling this information (operator 2). In each case, the failure occured during the operation of taking information. The mechanism of the error was confusion of address in the first case and omission of control in the second one (cf fig. 3).

The causes of confusion and omission had to be identified in studying (cf fig. 3) :

- the existing inadequacy between education/training and the task to be performed : operators did not really know the consequences of such errors on the safety of installation ; thus, they did not take care enough of the quality of their own actions
- the existing inadequacy between the work station design and the task to be executed : the first operator used his glove to write the address of the core element ; thus, a deformation of the address characters written on the glove, led to a reading error. The second operator had to stand up at each control because the information to control was behind him ; so, he did not stand up each time
- the influence of work organization : the operators were working in the same shift for the first time and they did not follow the written procedures ; thus, they did not use the same way of performing the task
- the influence of the history of the plant : for the first time, there was a human failure in refueling ; so, there was a decrease of operators' vigilance
- the physical environment : the second work station had a very hot environment
- the social environment : there is an isolation feeling during night shifts
- the time and duration of work : the failure occured during the first night of the shift, at 3 a.m., therefore in the worst conditions of vigilance.

Such an analysis give us a means to build a tree of causes for each human failure. This tree may be integrated to the simplified fault tree presented in chapter one. Then, all the information is formed in a unique tree.

CONCLUSION

By studying this unique tree and by cutting some branches (two branches for redundancy near the incident and only one branch at the other levels), we can find all the preventive actions to be taken.

This method is actually used in France to explain events that have happened and to prevent future incidents of the same kind.

In using this method for all representative incidents occured in nuclear plants, we hope building a data bank on human factors in order to understand the mechanisms of incidents due to human errors and to prevent these problems during design stage too.

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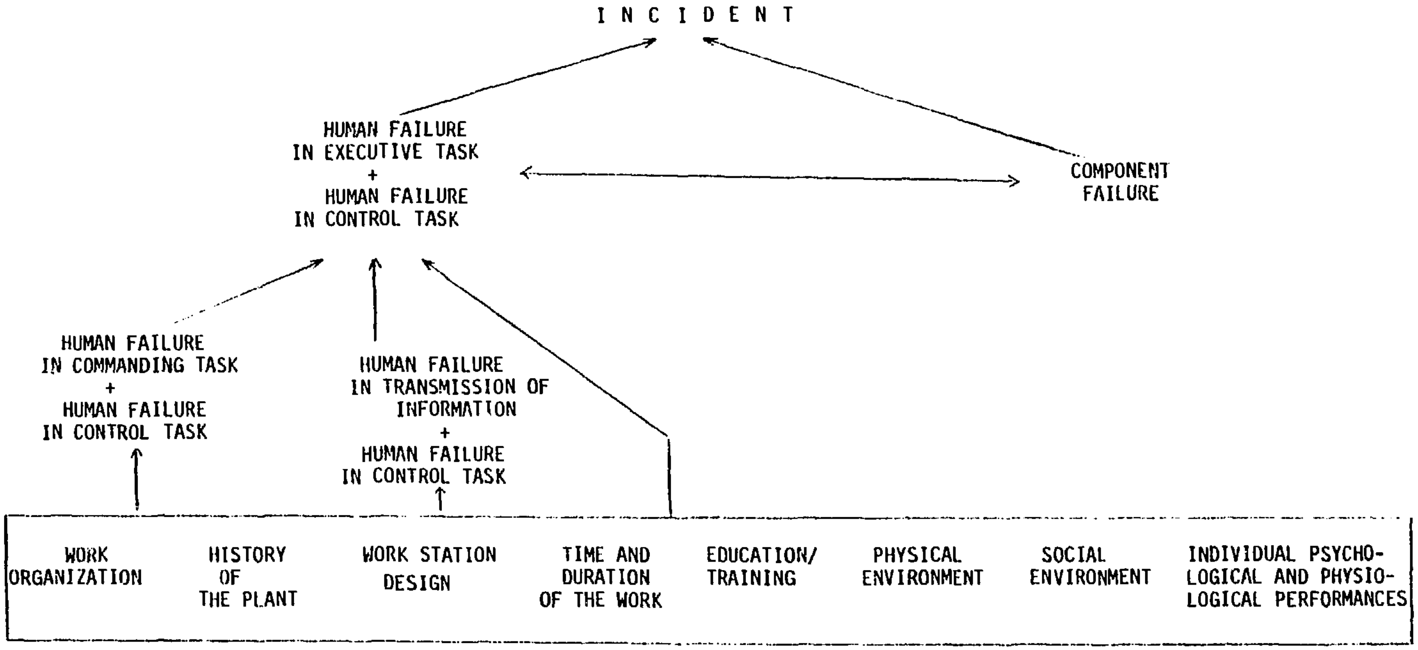

Figure 1 : DIFFERENT CAUSES OF AN INCIDENT

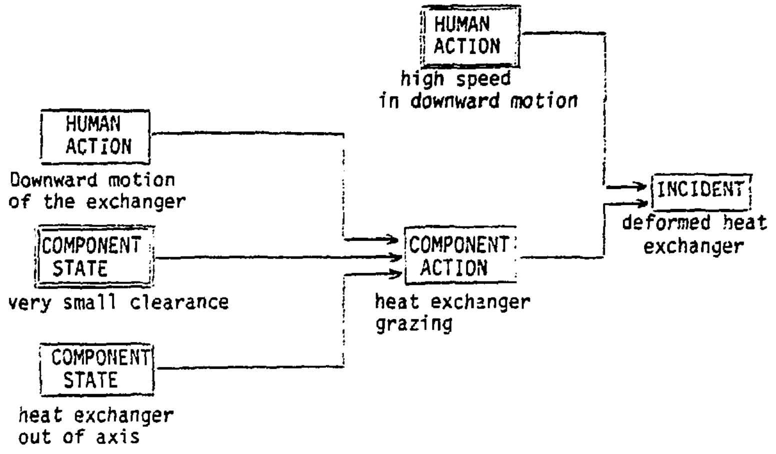

Figure 2.1. : EVENT SEQUENCE

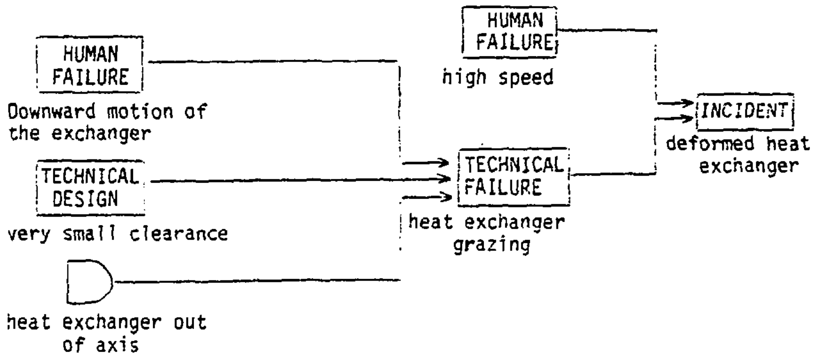

Figure 2.2. : FAULT TREE

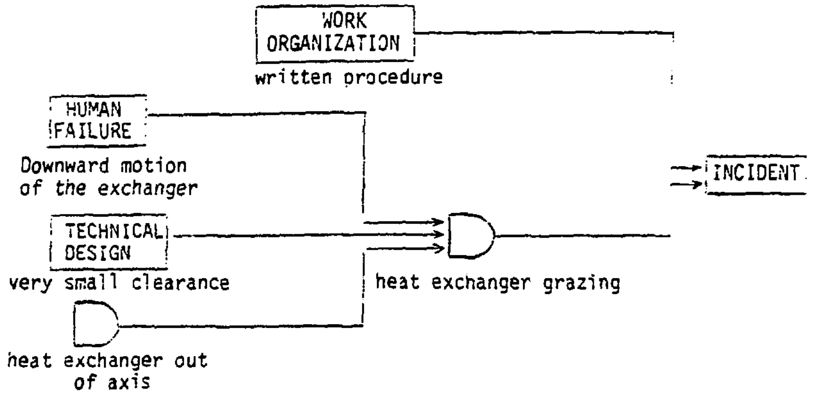

Figure 2.3. : SIMPLIFIED FAULT TREE

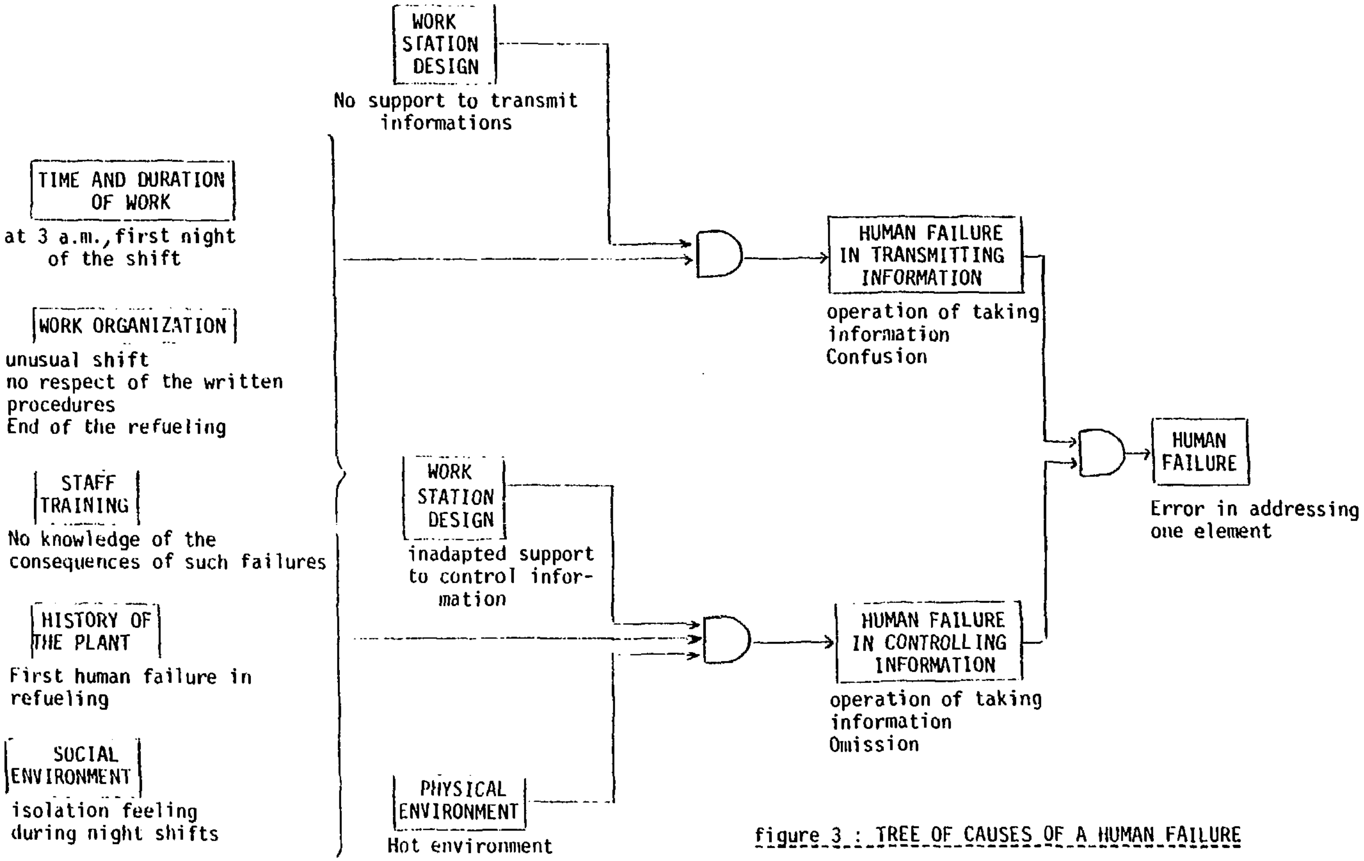

figure 3 : TREE OF CAUSES OF A HUMAN FAILURE

USE OF THE BIMODAL THEORY IN REACTOR CRISIS

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ABSTRACT

A model is proposed which explains distortions of perception and disruptions of analytical abilities under high stress. This is accomplished through a synthesis of three previously independent sets of observed phenomena: cognitive flexibility, behavioral conditioning, and loss of flexibility under stress. The model, the "Bimodal Paradigm", identifies two distinct modes of mental operation ("Actualizing" and "Programmed"), with stress as the modifying variable in the switch to rigid ("Programmed") operation. Implications are important and numerous, contradicting the interpretation of behavior under crisis as "erratic", providing specific indications for operator training and selection, control-panel design, and decisional procedures in crisis.

INTRODUCTION

It has long been observed, both in research and in common experience, that people tend to perceive according to previous expectations, that perception is often conformed to earlier impressions.[1,2] Thus, the performance of operators in nuclear facilities in crisis can become dangerously restricted to initial responses to the exclusion of subsequent information. In a like manner, behavioral analyses of factors contributing to operator error in crisis are generally limited to previous expectations, as we will see.

The Bimodal Paradigm offers a fresh perspective on human factors by identifying two mutually contradictory modes of mental operation, one flexible and one rigid.

The "Actualizing Mode" is described as flexible and aware. Access is high both to training and to hypothesis-testing abilities. The resulting behavior is situationally appropriate and intentionally chosen.

"Programmed Mode" operation is rigid, repetitive, and limited in awareness. In this mode, behavior is determined by previous stimulus-response conditioning which is instantly generalized to a present situation. Reaction time is very short. However, the generalized behavior pattern may or may not be situationally appropriate. A pattern which does not match the requirements of the situation will be rigidly repeated, once elicited in this mode of operation.

The effects of modal variability upon performance are considerable. The external conditions of modal modification have important implications in terms of operator selection and training, control-panel design, and decisional procedures during crisis.

THE FIRST IMPLICATION

A frequent impression of human performance in crisis situations is that the mind becomes erratic in operation.[3,4,5,6] The Bimodal Theory posits an alternative interpretation, that the minds of the operators in such situations are not erratic, but in the Programmed Mode, which is rigid and predictable. Mode "P" behavior may appear erratic to an external observer, because of the disruption of lateral thinking, including hypothesis-testing, and distortions in perception.[4,5,6] Actually, the behavior is rigid, locked onto one "track" of response. Accordingly, nearly all accounts of the LOCA at TMI-2 indicate the perseverance of one diagnosis of the malfunction, despite its failure.[7] In contrast, personnel fresh to the situation determined an accurate picture of the LOCA, including the open PORV.[7]

If predictable performance is dismissed as erratic, options become scarce for the prevention of recurrent errors in crisis. On the other hand, a Bimodal analysis allows for the anticipation of rigid behavior under stress, and indicates preventative measures for operator error.

THE MIND IN CRISIS

It is important to note that, because of the reduced awareness, the "switch" from the clarity of the Actualizing Mode to the rigidity of the Programmed Mode is usually imperceptible to the subject in crisis. Further, the causal variable in this unnoticed transformation into Mode P operation is high stress. Thus, in high stress, the mind loses the ability to think flexibly, to collect accurate data, and to identify and mobilize correct procedures. One may sense the adaptive features of such a shift in operation by considering the advantages of instant-response over cognitive clarity when confronted with a physical emergency, such as a saber-toothed tiger or a skidding automobile. However, if the wrong "programming" is mobilized, as happens when the most familiar response is a mismatch to the situation, flexibility becomes more important than the short reaction-time, high rigidity of the Programmed Mode.[8]

This Mode P behavior is not only rigid, but one may predict the direction it will take. As opposed to the awareness of multiple options which is characteristic of Mode A operation, awareness is restricted to whatever "track" (behavioral sequence) the mind first locks onto. This track will tend to be the most familiar, often the oldest, programming that can be associated with the situation.[9] Under high stress, thinking immediately becomes rigidly focused upon one interpretation of events. Contradictory information will be predictably and repetitively rejected,[10,11] even when such information is crucial. Hypothesis testing becomes impossible unless the individual has been trained to know that these predictable distortions may occur.

If the resulting Mode P behavioral "programming" happens to be appropriate to the situation, then it is judged to be intelligent and intentional. If the programming happens to be a mismatch with the requirements of the situation, it will appear to be erratic. Attempts to "control" such inappropriate behavior, once it has been judged to be erratic, will tend to produce greater rigidity. Furthermore, since there is a positive correlation between anxiety and Mode P operation,[12,13,14,15] as the behavior fails it will become more rigid, and produce further stress.

The tracking phenomenon found in crisis tends to be even more rigid in groups than in individuals.[5,16] Authoritarian leadership and dependency upon leadership are included in the rigidity.[5,17] Thus, people who feel that they bear the responsibility will "lead" the tracking, while the followers would not consider interfering with the leadership by offering their own observations of conflicting data.

REMEDIAL APPLICATIONS OF THE BIMODAL THEORY

We have noted that the conviction that the mind is erratic in crisis will inhibit effective measures to prevent tracking. Of even greater importance, when human factors specialists have sought the causes of human failure in reactor crisis, they have chosen between (1) personal deficiencies in operators and (2) lacks in programming (conditioning).[7] Such a perspective also excludes identification, and therefore treatment, of the effects of stress to track the mind onto one "program", to the exclusion of other training.

Fortunately, the authors have discovered repeatedly that external variables can be modified to facilitate Mode A performance during crisis.[18]

1. First, existing training can be revised to provide access, under stress, to training and procedures. For example, most people have experienced the "forgetting" of someone's name, only to later recall it. The name was, of course, not forgotten; only access was missing. For this reason, emergency procedures must be reorganized into formats that facilitate automatic access through associative chains-of-procedures, requiring neither thought nor feats of memory for availability.

Furthermore, it is a frequent accident of otherwise excellent training that mistakes become indelibly recorded in the trainee's programming along with correct procedures. In fact, the stress involved in mistakes suggests that the errors will become more prominent and accessible in the programming. Learning methods can be rearranged to prevent such contamination by accidental learning. For example, training can begin with completed procedures and back up a step at a time, an approach which is termed "Reinforcement by Results".

2. In addition, special operator training can be added in order to insure access, even in crisis, to both cognitive flexibility and emergency procedures. Central are: (1) training in the rigidifying effects of stress and in automatic stress-reduction skills; (2) training which provides familiarity with the Actualizing Mode, including practice with automatic "connectors" to these experiences; (3) "Regressed Association" is a process of eliciting rigid responses during training and establishing strong associations between each person's own predictable programming and the preferable training alternatives; (4) establishing "Interrupters" involves eliciting group-tracking responses during training and "attaching" program-interrupting phrases or images through repetition, such as phrases which incorporate exaggeration or humor.

3. As has been suggested,[7] control-panel design can be reorganized to reveal groupings according to function which would graphically represent

various systems of operation.

4. Involving supervisory personnel is recommended for some portion of training. For example, communication skills which are designed for crisis will counteract the usual freezing of information-flow under group-tracking.

Also, progressive desensitization[19] of operators to crisis is much indicated, with the eventual inclusion of supervisors to simulate interpersonal stress.

5. In contrast, the proposed creation of "Shift Technical Advisors" is contraindicated as presently designed.[20] The hierarchical rigidities of Mode P group-tracking would exalt the responsibilities of such individuals during crisis, with operators simultaneously disabled.

6. The selection of operators can incorporate Bimodal considerations. One may predict those most likely to refuse recognition of rigid response to emergencies. For example, those most certain that all of their behavior is intentional are the most likely to track "blindly".

THEORETICAL BACKGROUND

The Bimodal Paradigm synthesizes three previously independent sets of observed phenomena: cognitive flexibility, behavioral conditioning, and loss of flexibility under stress.

Mode P operation is familiar to anyone acquainted with either the classical or operant conditioning models predominant in Behavioristic Psychology. There is much support for the existence of this mode of operation, the major contemporary proponent of which is B. F. Skinner.[21,22,23]

Mode A operation will be recognized by those exposed to either cognitive "social-learning theory" or to Humanistic Psychology, again with both supplying considerable support for this mode of operation.[23,24,25,26,27]

Therefore, the uniqueness of the Bimodal Paradigm lies in the two-position analysis which identifies both types of operation in human behavior.[24] The existence of both positions is voluminously substantiated; the synthesis of both rigid and flexible factors of behavior into one model is a fresh perspective which allows for manipulation of modal factors.

While there has been no previous paradigm which has explained the rapid change from flexibility into rigidity, the observation that such rapid change occurs[12,13,14,15] is a longstanding one, and is well-popularized in psychology. In as well-known a text as <u>On Becoming a Person</u>, Carl Rogers cites a 1949 Ph.D. thesis from Columbia University in which Ernst Beier documented the significant rigidifying of abstract reasoning in students who had just received Rohrschach tests and diagnostic evaluations.[15] Equally, in perhaps the most standard introductory psychology sourcebook (Lindzey, Hall, & Manosevitz), a study is reprinted by Richard Barthol and Nani Ku which verified "regression under stress to first learned behavior."[9]

CONCLUSIONS

Using a Bimodal model of human performance, both distortions in perception and disruptions of analytical abilities become less perplexing. Inducers of rigid, Programmed Mode operation have been identified to include crisis, group-tracking, and hierarchical role-rigidity. All examples of stressful conditions, the above factors indicate modifications in operator selection and training, control-panel design, and decisional procedures during crisis. Effective approaches exist for access to training and clarity during crisis, but first require a recognition that the mind's first response to emergency is to abandon the flexible mode of operation.

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DIAGNOSTICS AT TMI USING NOISE ANALYSIS

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ABSTRACT

Noise analysis was used at Three Mile Island Unit 2 Nuclear Power Plant following the March 28, 1979, accident to assess the health of sensors, assure the absence of violent boiling, predict sensor failure, and others. In this paper, we consider the pressure noise from the primary pressure sensors located in the A and B loop. The primary objective in using pressure noise was as a degasification meter. From cross correlation analysis between the A and B loop pressure sensors, it was determined (a) the core barrel was still basically intact and moving in its normal pendulum mode, (b) there was not a large gas bubble located in the B loop, (c) both pressure sensors were in good health, and (d) the pressure noise was being driven by the reactor coolant pump which was exciting hydraulic resonances. When gas came out of solution, the hydraulic resonances (hence noise) were dampened. Therefore, the pressure noise was used as a degasification indicator from which operators were guided through the degasification process.

1. INTRODUCTION

At Three Mile Island Unit 2 (TMI-2) following the March 28, 1979, accident, noise analysis was used to assess the health of sensors, assure the absence of significant boiling, predict sensor failure, and others.[1,2] In this paper, the use of the noise from wide range pressure sensors to monitor degasification will be emphasized. Early on (∿2 days) into the accident, it was observed that the pressure noise from the A-Loop pressure sensor was high relative to that in the B-Loop. Furthermore, when the mean pressure was lowered slightly, the pressure noise decreased significantly, but reappeared when the mean pressure was increased again.

It was postulated that the hydrogen gas was absorbed in the water at the higher pressure and would simply come out of solution when the pressure was lower. It was postulated that the "spongy" system at lower pressure dampened the pressure fluctuations. The postulate seemed good, but there was uncertainty or confusion as to (a) what is the source of the pressure noise and (b) why is the pressure noise in Loop B much smaller than that in Loop A.

Considerable experience with pressure noise measurements at a sister B&W plant had been acquired by one of the authors (Robinson) at earlier dates. Furthermore, pressure noise had been used by Possa[3] for internal vibration measurements at TRINO PWR (ENEL). Using these past experiences, we were able to remove most of the uncertainty regarding the behavior of the pressure noise.

2. SPECTRAL ANALYSIS OF THE PRESSURE SIGNALS

Spectral and amplitude probability density (APD) analysis of A and B Loop pressure signals were performed with only the A-Loop primary coolant pump operating. Spectra were obtained at approximately 50 psi intervals during depressurization from 1000 to 300 psi (depressurization was performed to remove gas entrained in the primary water).

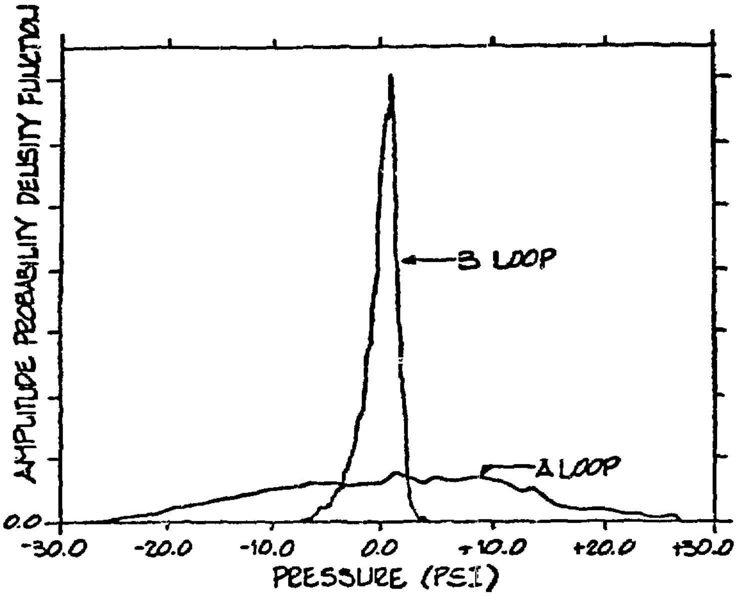
The pressure transmitters (Foxboro Model E11GH, 0-2500 psi) are located in the hot leg near the top of the primary piping leading to the steam generators. These transmitters have an estimated time response of ∿1.2 sec.

The pressure signals were obtained from a patch panel and conditioned for spectral analysis (using a Princeton Applied Research Model 113 preamplifier) by AC coupling to remove the DC component of the signal and amplifying the fluctuating signal.

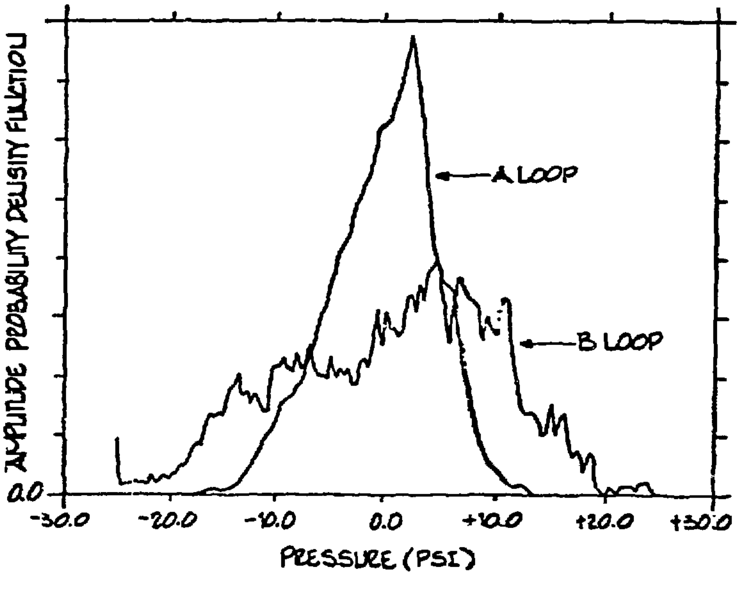
A Hewlett Packard Model 5420 dual channel spectrum analyzer was used to obtain probability density amplitude as well as simultaneous power spectra, cross power spectrum and coherence of the A and B Loop pressure signals. The analysis was performed over a range of 0-25 Hz with a resolution of ∿0.1 Hz. Approximately 1000 sec of data was taken at each pressure thus yielding an estimated standard deviation of ∿±10% on each spectral estimate. Approximately 76,000 samples were used for APD.

3. RESULTS OF NOISE INTERPRETATION

Representative results for A and B loop pressure fluctuations observed at high and low pressure are presented as APD functions in Figs. 1.a and 1.b, respectively. Note that the A-Loop pressure fluctuations are significantly larger than the B-Loop pressure fluctuations at high pressure. This caused concern that the B-Loop sensor might have failed (which would imply the A-Loop sensor also may be short lived) or that there was a large gas bubble in the B-Loop steam generator.



1.a Mean Pressure of 900 psi.



1.b Mean Pressure of 350 psi.

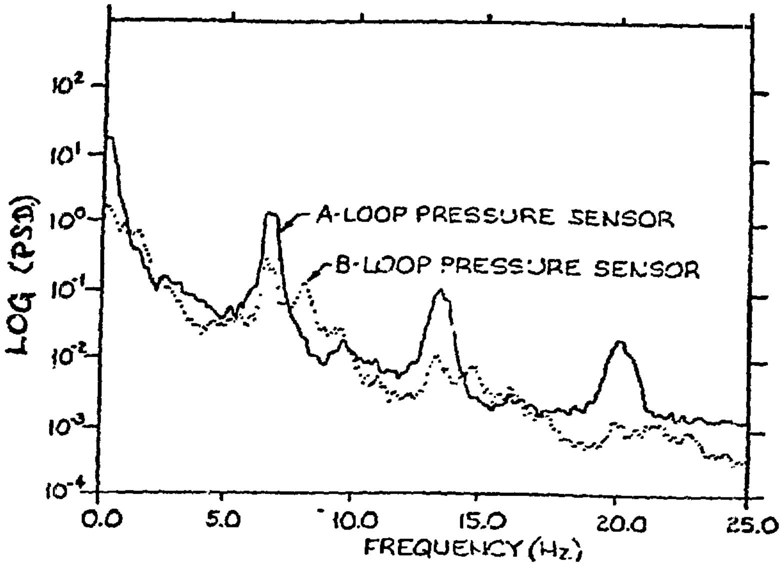
Figure 1. Amplitude Probability Density (APD) Function for Pressure Fluctuations from the A and B Loop Pressure Sensors at (a) 900 psi Mean Pressure and (b) 350 psi Mean Pressure.

To explore the concern with the A-Loop versus B-Loop pressure noise signals discussed above, extensive spectral analysis was carried out on these two sensors. Results obtained at a mean pressure of 950 psi are presented in Fig. 2. The significant features to be noted in Fig. 2 are:

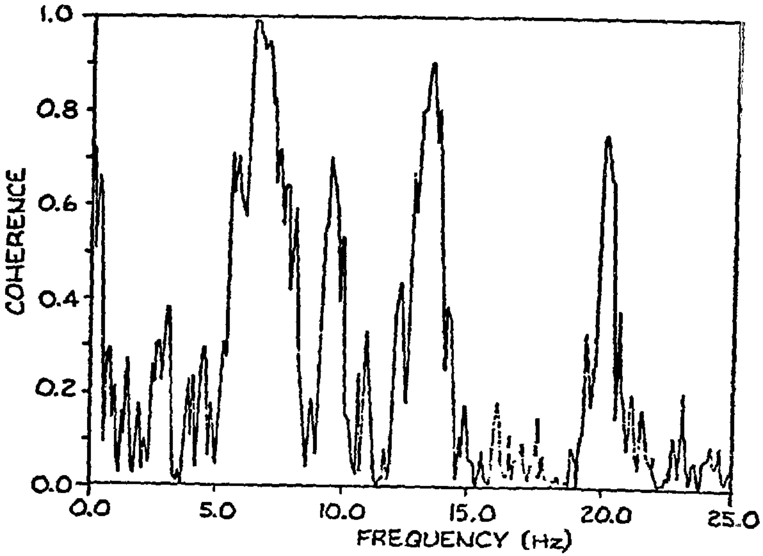
1. The low frequency activity in A-Loop is significantly greater than in the B-Loop
2. Relatively sharp peaks appear in the A-Loop spectra at 6-7 Hz and harmonics thereof
3. B-Loop has the same peaks at 6-7 Hz (and associated harmonics) but with lower magnitudes
4. B-Loop has peaks in the 8 Hz (and harmonics) frequency range which A-Loop does not have
5. The A-Loop and B-Loop spectra have peaks at ∿9 Hz frequency range of similar magnitude, but small relative to the 6-7 Hz activity (nevertheless significant as can be seen from the coherence at 9 Hz between these sensors in Fig. 2.b.)
6. The B-Loop has multiple peaks in the vicinity of the 2nd and 3rd harmonics of the dominant 6-7 Hz peaks of the A-Loop and B-Loop sensors

Most peaks observed in the spectra can be explained by postulating (a) the A-Loop has a resonant frequency in the 6 to 7 Hz band; (b) the B-Loop has a resonant frequency in the vicinity of 8 Hz; and (c) the A-Loop resonance is being excited which in turn is driving (modulating) the B-Loop. This situation would be expected to produce the multi-peaks (the sums and differences). This postulate checks out very well by carefully identifying the various peaks.

The only peak of significance not explained by the resonance/modulating argument is that one at ∿9 Hz. This peak is very similar to that observed by Possa[3] which had been identified as induced by core barrel motion in TRINO. Furthermore, this frequency is the normal frequency which has been identified in other PWRs as the expected core barrel motion. Therefore, we conclude that (a) there is still a core support barrel supporting a significant part of the core, (b) the pressure sensors are both healthy since the peak at 9 Hz is about the same in both sensors, and (c) there is not a gas bubble in the B-Loop steam generator suppressing pressure fluctuations in that loop (again concluded from the activity in both sensors at ∿9 Hz).



2.a Power Spectral Density (PSD).



2.b Coherence

Figure 2. Power Spectral Density (PSD) and Coherence for A-Loop and B-Loop Pressure Sensors at a Mean Pressure of 950 psi.

As postulated, the pressure noise should be significantly reduced upon lowering the pressure to a value not reached previously if dissolved gas is released at lower pressure. This postulate was confirmed when venting the gas from the primary loop caused the pressure noise to reappear. This effect was used to guide the operators in the degasification process.

Once the gas was removed, at a particular mean pressure, say P_o, cycling of the mean pressure at pressures in excess of P_o would always lead to repeatable noise behavior. The extremes of this behavior are presented in Fig. 3 for mean pressures of 1000 psi and 340 psi. The significant feature to be noted in Fig. 3 is that the resonant frequency at ∿8 Hz in the B-Loop decreases with a reduction in mean pressure. (In fact, the B-Loop noise becomes dominant as shown by the APD of Fig. 1.b).

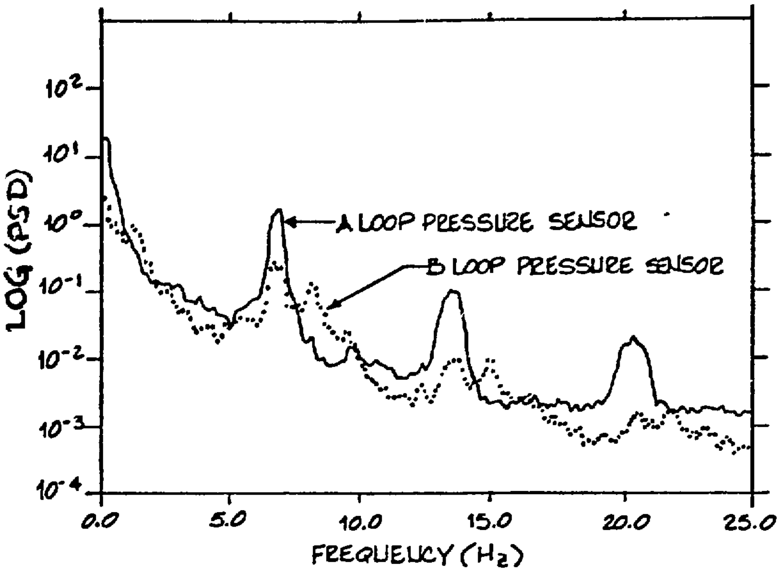
4. CONCLUSIONS

The noise from the pressure sensor signals was used to:

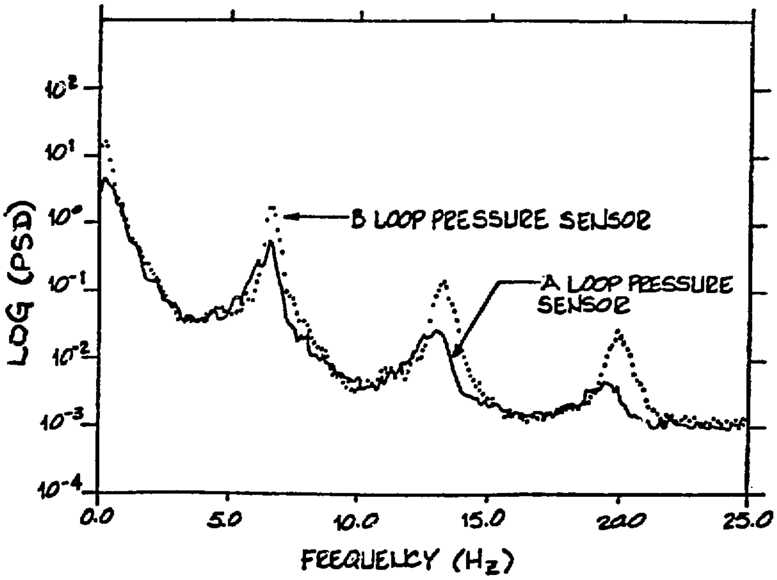
1. Monitor and guide the degasification of the primary loop.
2. Establish that the core support barrel was still exhibiting its normal pendulum motion.
3. Establish that both pressure sensors were healthy.
4. Establish that there was not a large gas bubble located in the B-Loop steam generator.
5. Determine that the source of the pressure noise was most probably a thermal/hydraulic primary system resonance excited by the pump.

Items 2 through 4 were established using experiences and data from other plants. Item 5 was conclusively established upon the inadvertent trip of the pump which was running, i.e., there was negligible pressure noise while the pump was off. Items 3 through 5 were important in establishing the validity of (and hence confidence in) the use of pressure noise as a degasification meter.

It should be pointed out that the pressure sensors were located in instrument lines coming off the exit coolant lines from the core (same was true at TRINO). In other nuclear plants, the pressure sensor may be located in sensor lines coming from the pressurizer. It should not be concluded that the noise characteristics for sensors monitoring the pressurizer would be the same as pressure sensors monitoring the coolant lines since in the former case the vapor state is being monitored, and in the latter case the liquid state.



3.a Mean Pressure of 1000 psi.



3.b Mean Pressure of 340 psi.

Figure 3. Power Spectra Density (PSD) for A-Loop and B-Loop Pressure Sensors at (a) 1000 psi and (b) 340 psi.

5. ACKNOWLEDGEMENT

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POST-ACCIDENT REACTOR ASSESSMENT BY DYNAMIC MEASUREMENTS

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ABSTRACT

During the thirty days following the initiating event of the Three Mile Island Unit 2 accident, vital information to assess the condition of the degraded primary system was acquired by means of dynamic measurements. Three areas, out of the numerous dynamic measurements performed, are discussed: structural-mechanical evaluation of the primary system, non-condensible entrained gas volume calculations ("bubble size"), and primary system instrumentation degradation monitoring. In each of these areas, only through the techniques used could the needed information of the status of the primary system be reliably acquired.

INTRODUCTION

Approximately sixteen hours after the initiating event of March 28, 1979, an interim reactor core cooling mode was established at Three Mile Island Unit 2. The reactor coolant system pressure was being maintained between 1000 psig and 1100 psig with heat being removed from the reactor core using the steam generator A and forced circulation. The TMI-2 accident had entered the post-accident phase. Efforts were immediately initiated to assess the structural and mechanical condition of the primary system, remove entrained gas from the primary system, and monitor the stability and parametric state of the system, in order to achieve cold shutdown conditions. At 1408 hours on April 27th, the last operating reactor coolant pump was deactivated and natural circulation was achieved. The TMI-2 primary system was finally in a cold shutdown stable condition and had entered the plant recovery phase.

During the post-accident phase, dynamic measurements, using specialized analysis equipment, were performed on a variety of signals emanating from the inaccessible containment building. These dynamic measurements provided vital information in accessing the the condition of the degraded primary system. This paper addresses the dynamic measurements methods used in three areas: structural-mechanical evaluation of the primary system, non-condensible entrained gas volume calculations ("bubble size"), and primary system instrumentation degradation. During this period numerous other state-of-the-art dynamic measurement were performed with signals from in-core thermocouples, ex-core ion chambers, in-core self-powered neutron detectors, and a host of plant parameters (Ref. 1, 2, 3, and 4).

STRUCTURAL-MECHANICAL EVALUATION

The structural-mechanical evaluation consisted of determining the effects of continued operation of the degraded primary system in the forced circulation mode. The principle concerns were the possibility of seriously damaging the operable steam generator with impacting loose parts and the possibility of altering the integrity of the core. Additionally, indications of the structural condition of the reactor intervals were sought. The installed Loose Parts Monitoring system sensors (accelerometers) and special high-resolution analysis equipment were used.

A number of small magnitude metal-to-metal impacts were detected and analyzed. These emanated either from the inlet to each steam generator or the bottom of the reactor vessel. No special significance was associated with these impacts due to the small energy content and they were postulated to be small debris which would not jeopardize the structural integrity of the NSS. No indication of severe structural-mechanical degradation was observed. Additionally, by correlating reactor vessel accelerometer signals with primary pressure fluctuations, it was determined that the beam mode frequency of the reactor internals had not varied substantially from the established baseline. This indicated that the internal structures were not severely damaged.

NON-CONDENSIBLE ENTRAINED GAS VOLUME CALCULATIONS

During the accident phase, the reactor core temperatures became sufficiently high to cause a chemical reaction between the zircaloy metal that clads the fuel elements and water which generated non-condensible gases, mainly hydrogen. The main concern was to maintain the reactor core covered with water and establish a method to vent the non-condensible gases from the primary system.

A first order gas volume calculation method was devised to monitor the volume of non-condensible gases. These calculations were referred to as the "bubble size" calculations and were based on the following expression:

$$V_B = \frac{P_2 \, \Delta V_{Total}}{P_1 - P_2}$$

where: V_B - bubble volume in ft^3 at RCS pressure P_1

P_1 - RCS pressure at the start of the primary pressure decrease

P_2 - RCS pressure at the end of the primary pressure decrease

ΔV_{Total} - sum of the pressurizer and make-up volume changes during the RCS pressure decrease

The procedure consisted of decreasing the RCS pressure by 100 psi and monitoring the changes in level of the pressurizer and make-up tank. This method was successfully used for the initial "bubble size" determination. However, as the "bubble size" was decreased by the pressurizer venting technique, significant variations in the estimated "bubble size" occurred which indicated that perhaps the gas venting was not being accomplished (Ref. 5). The start of the variations were correlated with the start of severe primary systems pressure oscillations. Figure 1 illustrates an eight second time segment of RCS pressure fluctuations which exhibit a 40 psig peak-to-peak fluctuation. Since the data used in the "bubble size" calculations were based on instantaneous values obtained from the plant computer, significant errors were introduced. By use of dynamic measurement techniques, the calculations performed were once again consistent and "bubble size" calculations were performed in the presence of substantial pressure fluctuations.

SENSOR FAILURE MONITORING

A program to monitor incipient sensor failures was initiated when it was recognized that the reactor building radiation levels had led to exposure exceeding the radiation limits of some key primary system sensors. This program was based on noise analysis methods. The sensors in question included the reactor coolant pressure, pressurizer level, and steam generator level.

One of three redundant pressurizer level sensors failed within a week after the accident. Periodic power spectral density measurements show that this failure was accompanied by a substantial increase in signal noise and a corresponding loss of spectral features. Continued monitoring revealed a period of erratic behavior of the number two pressurizer level signal. This was indicated by a large increase in signal noise and loss of spectral features along with sudden DC shifts. These indications cleared and no further problems were observed for about one week. Intermittent spikes and step changes in the DC level were then observed on the chart recorder over a period of about ten hours. The plant personnel were alerted to the potential imminent failure of this sensor which occurred within five hours. This monitoring program continued through the transition to natural circulation with minor sensor degradation observed which was not judged to lead to sensor failure.

CONCLUSION

Sixteen hours after the initiating event at Three Mile Island Unit 2, a transient dynamic interim core cooling mode was established. Thirty days later, a steady-state cooling mode was finally achieved. During these thirty days information gained by use of dynamic measurement techniques was vital in the assessment of the condition of the reactor coolant system. In several instances, only by these measurement techniques could the needed information be acquired. Dynamic measurements must be considered as part of the methods used in future post-accident reactor assessments.

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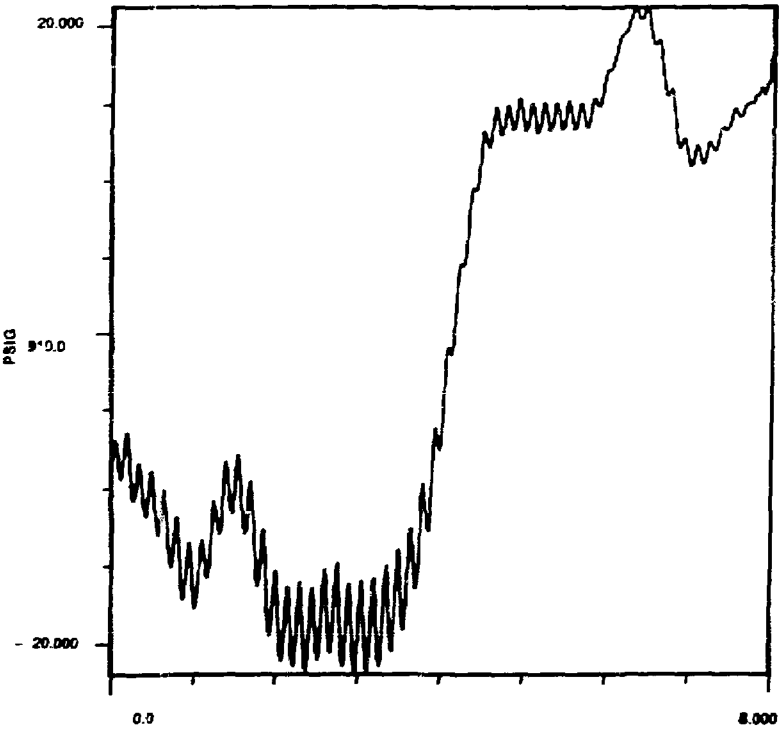

FIGURE 1. REACTOR COOLANT SYSTEM PRESSURE FLUCTUATIONS (RCS PRESSURE: 910 PSIG)

Dup

EXPERIMENTS ON HYDROGEN FOR THREE MILE ISLAND

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ABSTRACT

Experiments were conducted in non-radiation environments during the Three Mile Island accident to investigate hydrogen behavior in the system and to evaluate potential methods for hydrogen scavenging from the system. Equilibrium solubility and the dissolution behavior of hydrogen in water was investigated. A pressure rebound occurred after a step depressurization.

Absorption of hydrogen into hollow glass microsphere was found to be too slow to control a hydrogen evolution. Chemical reactants which would control hydrogen were not found. Chemical scavenging of hydrogen in water by catalyst introduction in the presence of dissolved oxygen identified a platinum catalyst as effective.

INTRODUCTION

Starting on April 1, 1979, Billings Energy Corporation under the direction of EG&G Idaho, Inc., undertook a series of tests for the Nuclear Regulatory Commission to provide information regarding (1) potential amount of hydrogen in the primary coolant water in the Three Mile Island 2 Reactor; (2) methods of scavenging gaseous hydrogen from the reactor system; and (3) the determination of the most efficient and also the safest means of depressurization. Although only small amounts of hydrogen were later found in the system, this study produced information of interest for similar accidents in which hydrogen remains in the system. No investigation of radiochemical effects was made; the study focused on non-radiation solubility and chemical effects. Most of the work was completed within one week of its start.

SOLUBILITY OF HYDROGEN IN WATER

Calculations were made to determine the maximum amount of gaseous hydrogen that could theoretically be dissolved in water at various temperatures and pressures. These computations gave an indication of the potential hydrogen bubble growth in the reactor during depressurization. It was estimated that the maximum bubble growth in the Three Mile Island Reactor would be 1037 cubic feet at 300 psia (9% of the total volume), assuming the water in the reactor was completely saturated with hydrogen gas. A series of bench-scale tests and a larger, pilot-plant-scale test were performed to verify these calculations and to examine actual hydrogen behavior.

A one-liter test apparatus was used to investigate the solubility of hydrogen in water under conditions analogous to the Three Mile Island Reactor. Depressurizations of the test unit were performed for two cases: (1) continuous discharge, and (2) fast discharge with intermediate stops. Although the results were higher than the theoretical calculations (a bubble growth of 1781 cubic feet, 15.5% of total, and 1884 cubic feet, 16.4% of total, for the Continuous Discharge Test and the Fast Discharge with Intermediate Stops Test, respectively), the bench scale tests established a base for the Reactor Simulation Tests, as well as aiding in the definition of the experimental testing procedure.

Using the bench scale test apparatus, the hydrogen bubble growth during depressurization was experimentally determined as a function of time at a constant temperature. A "rebound effect" was observed in these tests. That is, a sudden drop in pressure would be followed by pressure rise which erased 10 to 90% of the original drop as shown in Figure 1. The rebound was a function of the pressure at the start of the step, as shown in Figure 2. The results of these tests showed that a large pressure rebound could be an indication of saturation.

HYDROGEN SCAVENGING

Catalytic System. Several different schemes were considered for catalytically causing the gaseous hydrogen in the reactor to react with oxygen to form water and thus reduce the pressure in the reactor vessel as well as reduce the hydrogen gas volume.

The first consideration was to analyze the materials in the reactor vessel itself and look at the possibility of these materials catalyzing a reaction between the gaseous hydrogen and oxygen. It was concluded that, although possible, it would be highly unlikely that significant reaction would take place. Radiation recombination of hydrogen and oxygen could of course take place with proper concentrations.

A literature search identified some substances that could possibly be used to catalyze this reaction. This survey indicated that hydrogen forms complexes such as ReH_4^-, $HCo(CN)_5$ and $HPtBr[P(C_2H_5)_3]_2$. Four other materials that would possibly catalyze the reaction of hydrogen with oxygen are: (1) colloidal dispersion of sodium boronhydride reduced nickel (or platinum); (2) a finely ground alumina-supported nickel (or platinum); (3) a homogeneous $Co(CN)_5^{3-}$ complex; and (4) catalyst coated glass microspheres.

To define the identified catalysts more adequately, several experiments were conducted.

Of the several catalyst systems considered, the catalytic reduction of oxygen with hydrogen under water on a platinum catalyst showed positive experimental results. Total pressure reduction in a one-liter hydrogen-air-water system over one hour's time indicated nearly complete hydrogen-oxygen reaction.

Microsphere Scavenging. Another alternative that was considered which might reduce the amount of hydrogen in the reactor vessel was the possibility of introducing hollow glass microspheres to the circulating water system. The ability of hydrogen to diffuse through the wall of the microspheres indicated that they would have a hydrogen scavenging effect. The experiments on the bench scale test apparatus indicated that there was a 10.6% decrease in pressure. The results did indicate that although the microsphere scavenging effect would not be an immediate solution to depressurization, the long term effects of the microspheres would have a significant result on the hydrogen partial pressure of the system.

Hydrogen Peroxide Scavenging. An experiment evaluated feasibility of introducing hydrogen peroxide into the reactor system to react with hydrogen.

It indicated that without radiation, the decomposition of the peroxide takes place at a much more rapid rate than the combining of the oxygen and hydrogen in the system. The results showed a net increase in pressure rather than a decrease in pressure.

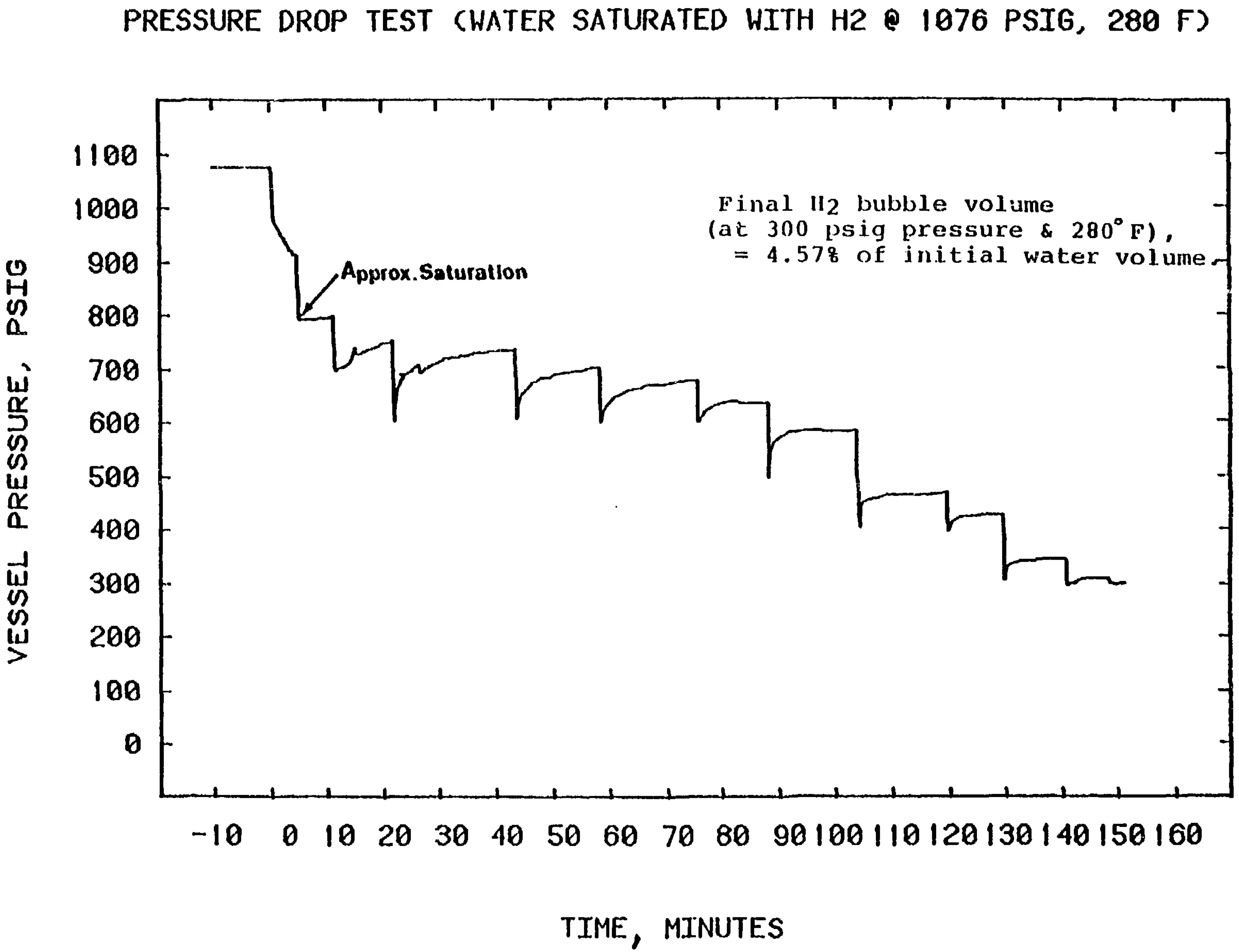
REACTOR SIMULATION TESTS

A pilot plant unit was constructed to simulate the reactor at Three Mile Island. The simulated reactor system consisted of the following major components: the steam generator volume, the high pressure circulation pump, the reactor vessel volume, and the pressurizer vessel. The pilot plant was instrumented to allow the monitoring of temperatures, pressures, flow rates, and volumes. Figure 3 is a schematic diagram of the simulated system.

The purposes of the reactor simulator tests were: (1) to determine the effects of pressure and temperature reductions of the reactor system filled with water containing various amounts of hydrogen up to saturation; (2) to obtain a model of these characteristics so that the degree of hydrogen saturation and bubble size might be ascertained through pressure and temperature and (3) to determine the best method for cold shutdown of a nuclear reactor system believed to contain a hydrogen water solution.

The tests confirmed the behavior shown in the bench scale tests, that is, solubility and the behavior during depressurization (see Figures 4 and 5). The pressure rebound effect was found to be a good indication of the saturation pressure of the dissolved gases. When the logarithm of the pressure was plotted against hydrogen bubble size (see Figure 6), the data points fell on a straight line which, when extrapolated to zero bubble size, gave the initial saturation pressure of hydrogen. Concentration difference effects became evident in these tests. Low flow past the heater in the reactor vessel in one test produced substantial degassing of the water. Re-solution of this gas was much slower than the degassing process. A similar effect in heated low-flow portions of reactors may be a major contributor to gas pockets in shutdown reactor systems and should be accounted for in evaluations of gas behavior in accidents such as Three Mile Island.

It was concluded that the most efficient and safest procedure for the cold shutdown was first to reduce the temperature of the reactor system. A temperature decrease reduces the amount of hydrogen in solution, and hydrogen evolves from the water. Since unsaturated water is added to the system during this period the actual amount of gaseous hydrogen does not increase significantly. After the temperature of the system has been decreased, the pressure and flow rate can then be reduced.



Final H_2 bubble volume
(at 300 psig pressure & 280° F),
= 4.57% of initial water volume

Figure 1 Pressure - Time Trace Showing Changes and Subsequent Pressure Recovery as Hydrogen Comes Out of Solution.

WATER SATURATED WITH HYDROGEN @

1076 psig, 280°F

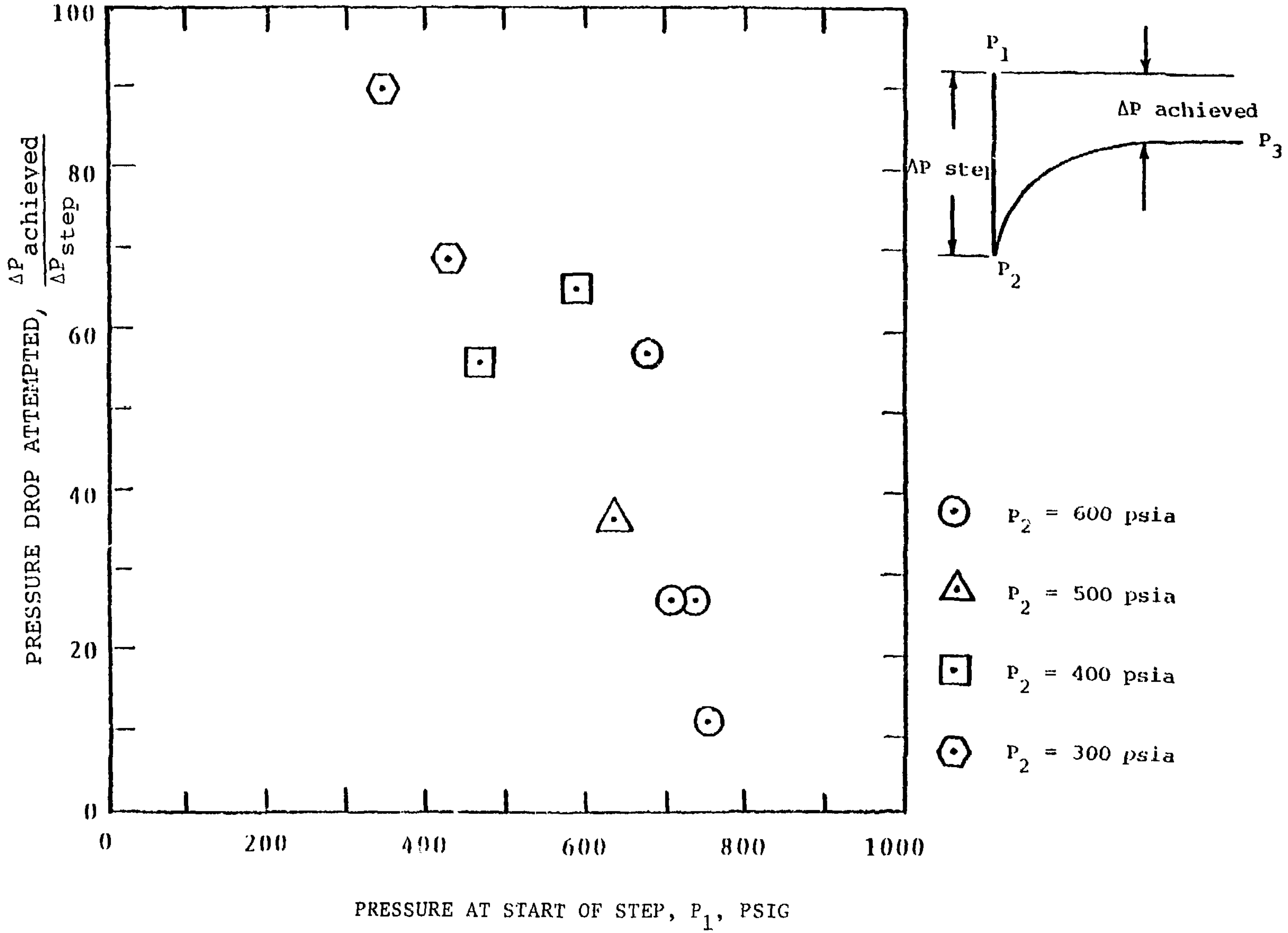


Figure 2 Interpretation of Data in Figure 1

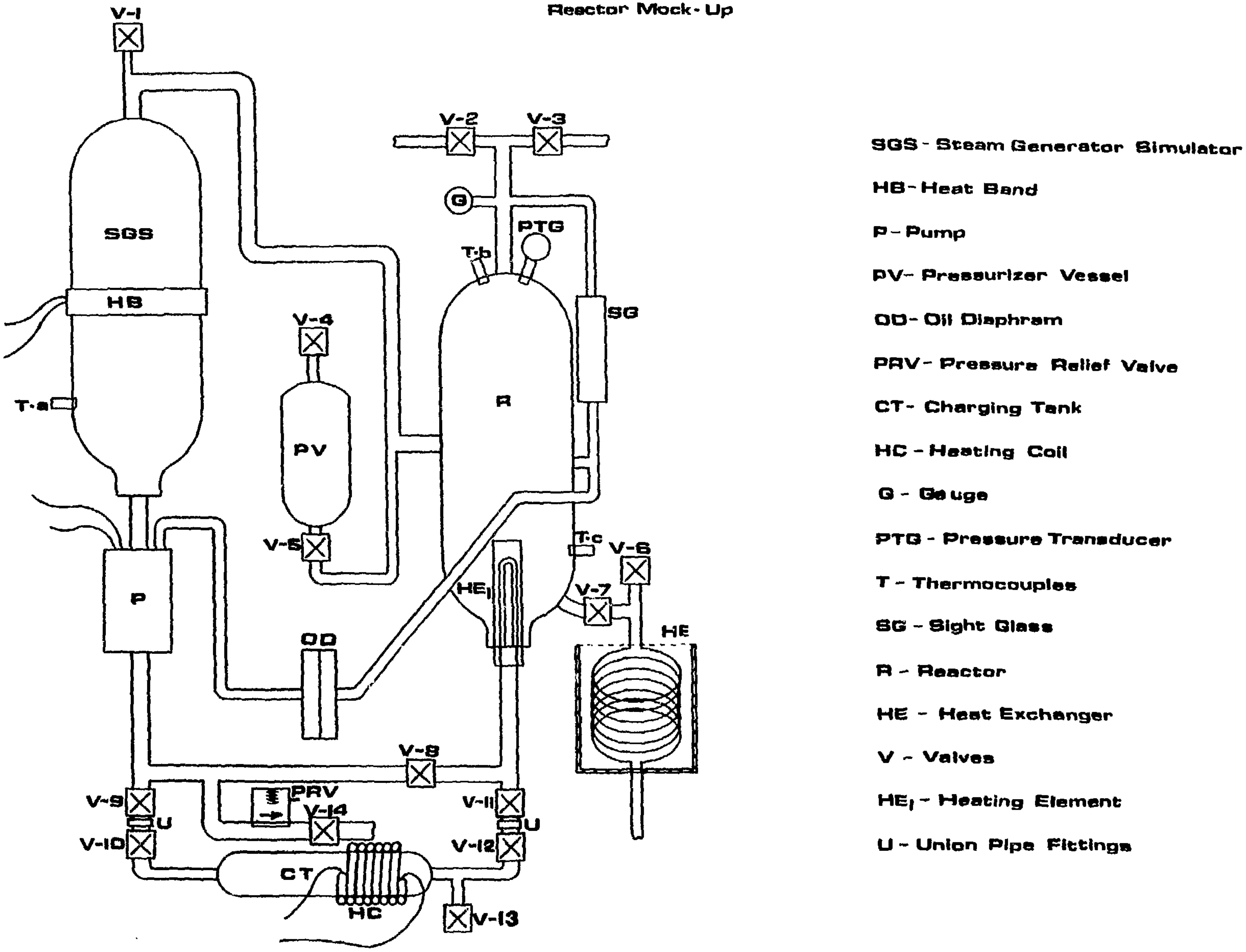


Figure 3 Schematic of the Simulated Reactor Apparatus

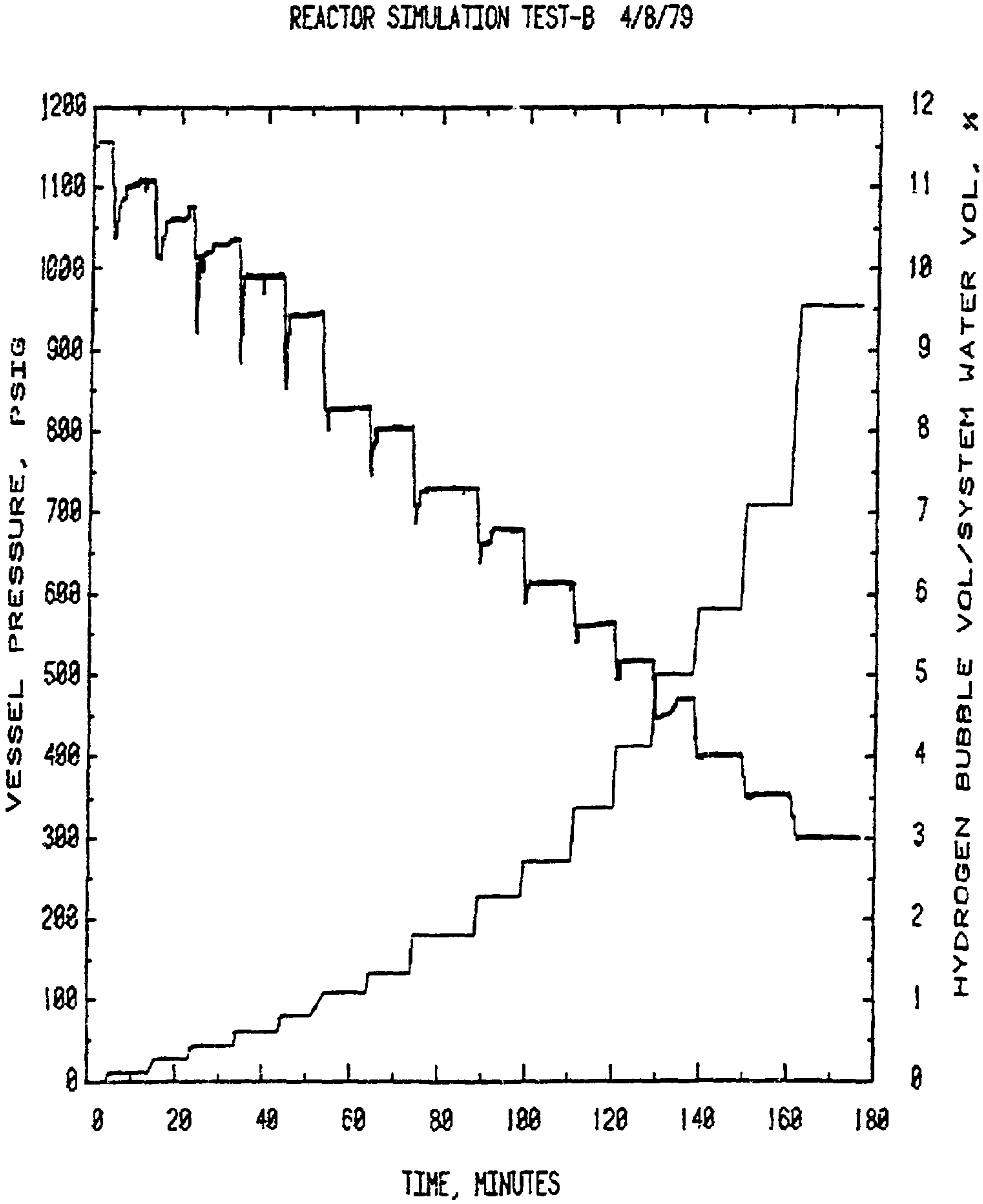


Figure 4 Plot of Hydrogen Bubble Growth vs. Pressure, Test B

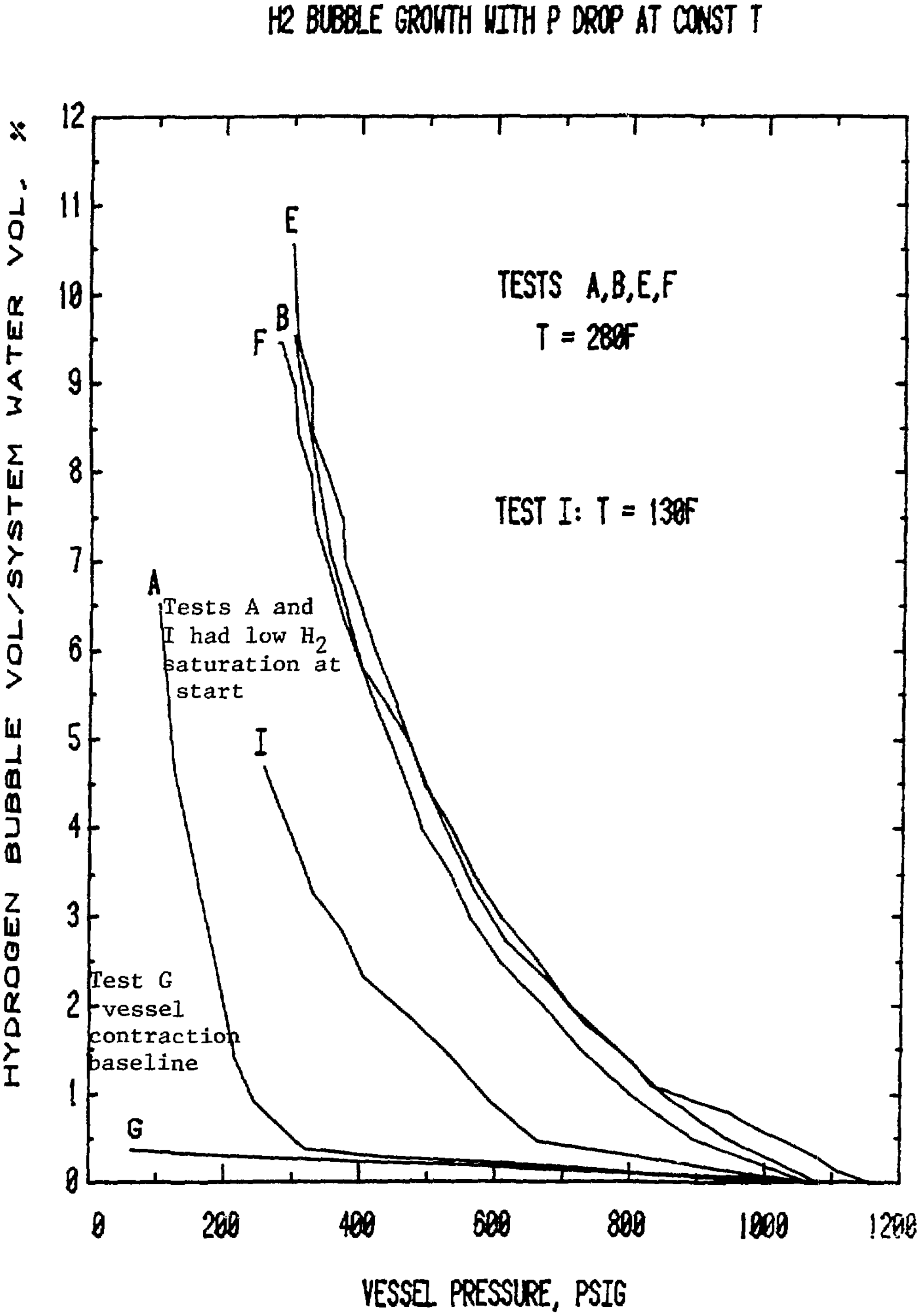


Figure 5 Composite Plot of Tests A, B, E, F, G, I Showing Bubble Growth as a Function of Pressure

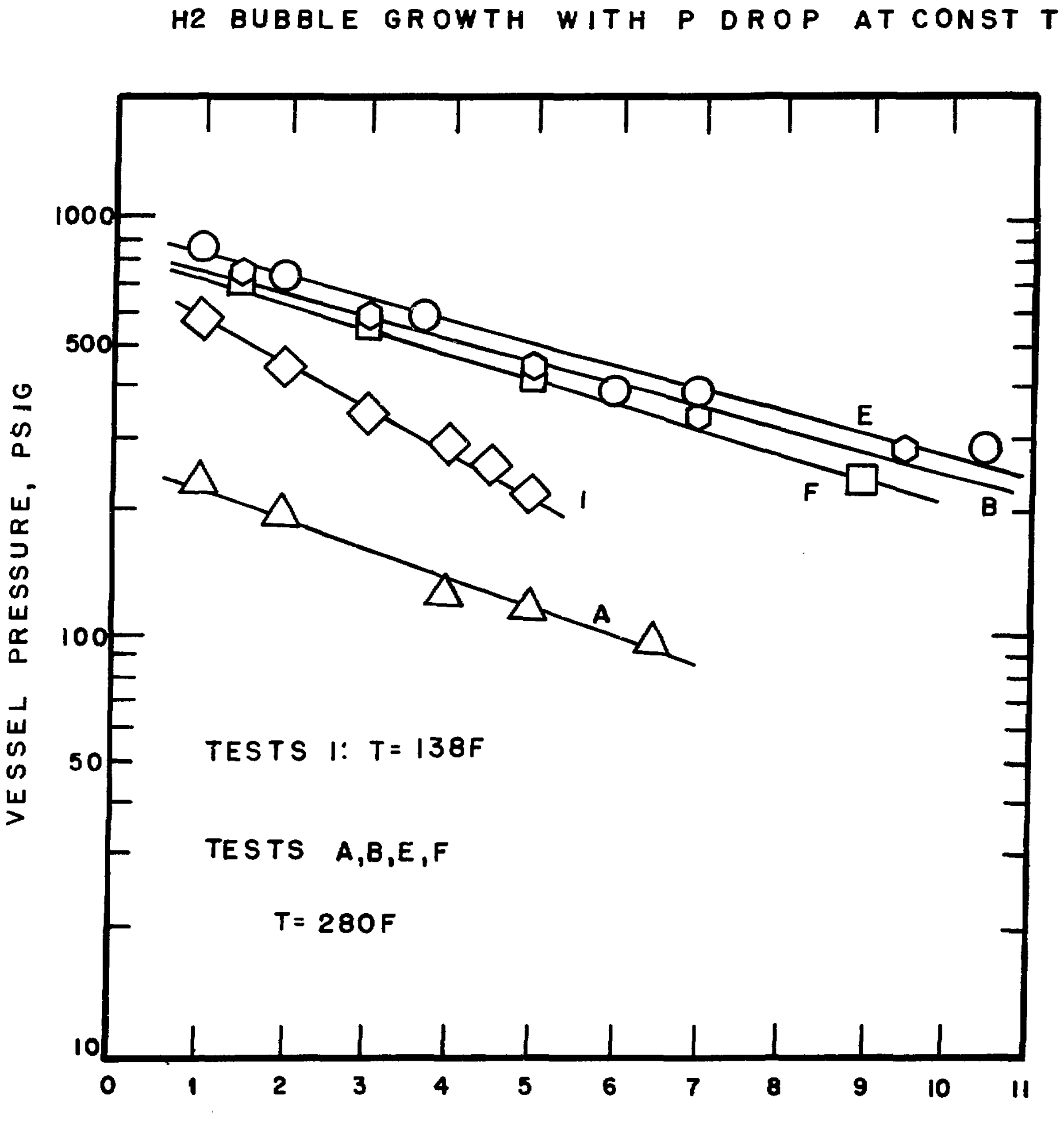


Figure 6

Composite Plot on Semi Log Paper of Bubble Growth as a Function of Pressure, Tests A,B,E,F,I

THE THREE MILE ISLAND UNIT 2 (TMI-2) CONTAINMENT ASSESSMENT TASK FORCE PROGRAM

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ABSTRACT

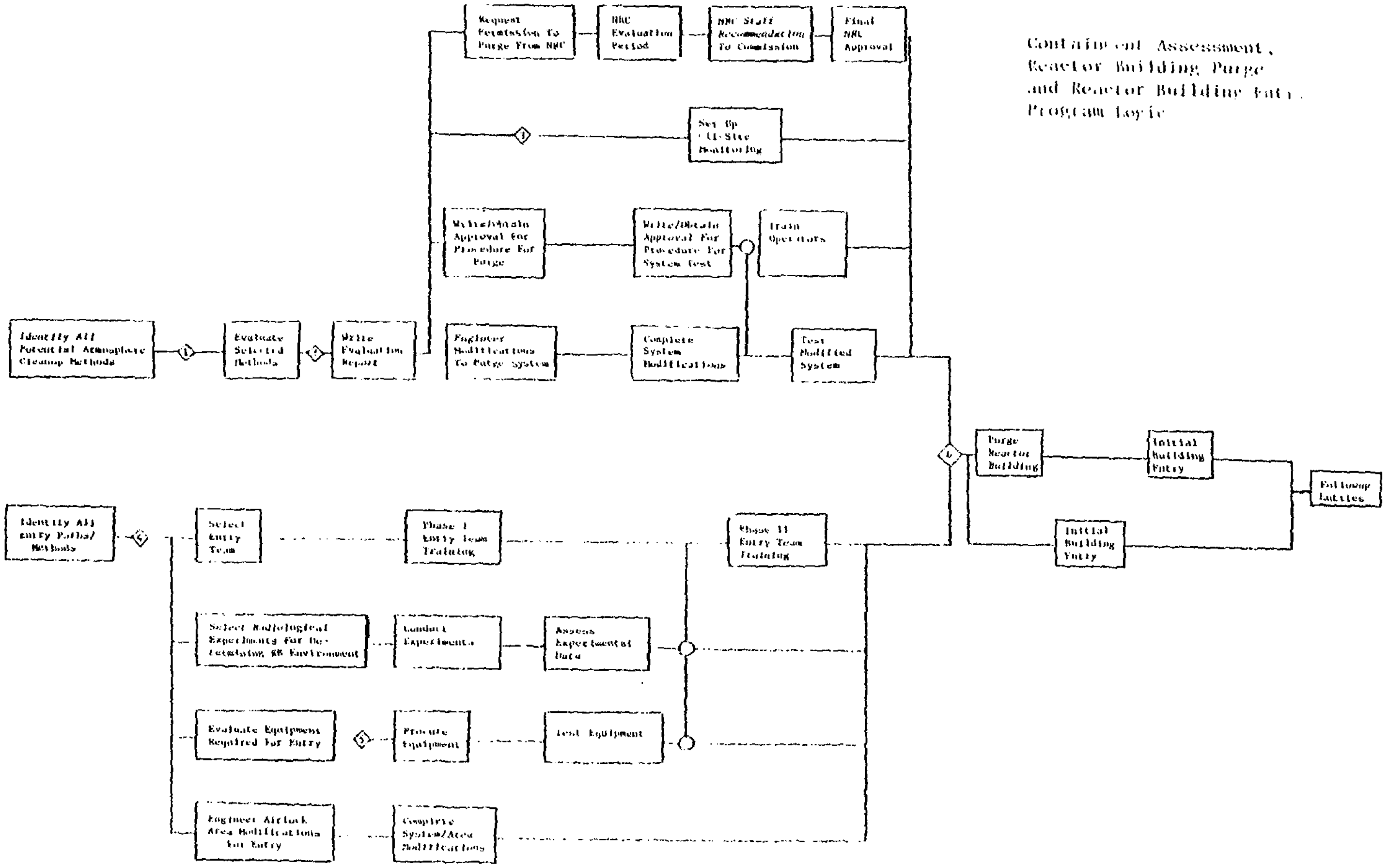
Following the accident at TMI-2, General Public Utilities/Metropolitan Edison established a Containment Assessment Task Force (CATF) to assess the radiological and physical status of the containment through a planned program of experiments and use those data to plan for initial entry into the containment, containment atmosphere purge, and containment recovery. The experiments to date have included airborne samples, gamma radiation readings through the equipment hatch, gamma radiation readings through two different penetrations, radiation mapping and entry into a containment airlock, a sump water sample, and insertion of a TV camera and radiation monitors through a penetration. These experiments have shown that general area radiation levels are in the range of 1.5-3 rem/hour, which is significantly less than originally calculated. The CATF program has been a successful program which has contributed valuable information to the TMI-2 recovery effort.

INTRODUCTION

Following the March 28, 1979 incident at Three Mile Island, Metropolitan Edison set up a Containment Assessment Task Force. The overall goal and objective of the Containment Assessment Task Force was to establish, using external measurements, the reentry environment inside the Reactor Building. The specific function of the Assessment Task Force was to gather data as necessary to ensure that containment atmosphere cleanup could be accomplished, that an initial entry into the building could be accomplished, and that planning for Reactor Building decontamination and recovery could proceed. The cleanup of the containment atmosphere and the initial entry were considered to be the initial vital steps in beginning the recovery of the Three Mile Island Unit 2 Containment Building. The entry was also specifically required in order to gather additional data by visual inspection and detailed radiological surveys to support planning for the decontamination of the Reactor Building.

In order to accomplish the assigned duties, the Containment Assessment Task Force established a number of experiments and measurements which could be run without benefit of building entry. These experiments were designed to determine as much as possible what the airborne contamination, contamination plated out on the various surfaces of the containment, and contamination contained in the water in the basement of the Reactor Building were. The experiments and measurements were specifically designed to obtain detailed technical data as much as possible on the magnitude, identity, distribution, and chemical forms of the existing airborne, surface, and sump water activity and the resulting radiation levels. The final step of the Containment Assessment Task Force would be the initial building entry in order to take direct radiation surveys, obtain material for decontamination studies and perform a preliminary visual assessment of damage within the building. The logic behind the assessment, purge, and reentry programs is shown in Figure 1.

Containment Assessment,
Reactor Building Purge
and Reactor Building Entry
Program Logic



Decisions:

① Eliminate methods unproven at high (>500 cfm) flow rates and/or not commercially available.

② Select best cleanup method (purge). Choose detailed method of conducting the purge. Determine the modifications required to make the purge system ready for use.

③ Determine requirements and capabilities.

④ Select air lock to be used for entry.

⑤ Select communications equipment, camera, clothing, breathing equipment, lighting, and radiation/dose monitors.

⑥ Determine feasibility of building entry prior to purge and likelihood of obtaining permission to perform the purge in the near term.

The measurements and experiments taken as part of this assessment program included the following:

1. Weekly containment building airborne samples. These samples were analyzed for particulates, gases, iodine and gross beta.

2. Gamma radiation readings through the equipment hatch, using a Ge(Li) detector. The purpose of these measurements was to determine the isotopic identity and magnitude of plateout on the 305' elevation.

3. Gamma radiation readings through the inner flange of penetration R605 (approximately 2 feet above the sump water level, near the basement of the Reactor Building) using a Ge(Li) detector and a teletector. The purpose of this measurement was to determine sump level and specific activity on the contamination in the sump.

4. A sump water sample. In order to perform this sampling, a hole was cut in the inner flange of penetration R401 (approximately 2 feet above the sump water level) and water was drawn in order to accomplish a detailed activity analysis of the water. Subsequently, several larger samples were drawn for further analysis.

5. Gamma radiation readings through the inner metal flange of penetration R626 (at the 347' elevation approximately 11 feet above the Reactor Building operating floor) using a NaI(Tl) detector and teletector. The purpose of this measurement was to determine general area radiation levels and to determine the isotopic identity and magnitude of plateout on the 347' elevation operating floor.

6. Radiation mapping of the number 2 personnel air lock. The experiment consisted of taking air samples from the personnel air lock and also placing probes into the air lock to determine airborne activity radiation level inside the air lock.

7. Analysis of the hydrogen recombiner inlet spool piece. This experiment consisted of removal of the spool piece to the recombiner and shipment of the spool piece to Oak Ridge for analysis. The purpose of the experiment was to determine what plateout existed on the spool piece as a result of the several days of flow through the hydrogen recombiner which occurred within the first three weeks after the accident.

8. Remote TV camera and radiation surveys through penetration R626. The purpose of this experiment was to obtain an initial visual assessment of the damage that may have been done by the accident and to obtain the first direct radiation measurement readings inside the building.

9. Air lock entry. This experiment consisted of opening the outer door and entering the air lock in order to take detailed swipe surveys, radiation surveys and Ge(Li) scans through the inner door of the air lock. The purpose of this experiment was to obtain better information on the 305' elevation radiation levels and the 305' elevation plateout source. The experiment was also expected to afford some view through the inner door viewport of the 305' elevation.

With the exception of the sump sample, the above experiments were all taken by the Containment Assessment Task Force as part of the initial entry program. The sump sample was actually taken in order to better define activity levels in the sump in order to plan for the initial engineering of a sump water cleanup system. Each of the above experiments are described in general and the results of each are presented in this paper. Three of the above experiments will be described in greater detail in other papers presented at this conference. These experiments are gamma radiation readings through the equipment hatch, gamma radiation readings through penetration R605, and gamma radiation readings through penetration R626. Additionally, the remote TV camera operations and radiation surveys through penetration R626 will be described in greater detail at the annual ANS meeting in Las Vegas in June.

WEEKLY AIR SAMPLE PROGRAM

Samples of the Reactor Building atmosphere have been taken and analyzed routinely since the March 28th incident. Initially, airborne activity samples were difficult to obtain due to the high radiation levels of the gases and also due to the fact that the normal sample panel was in the auxiliary building, where high general area radiation levels existed. As a result, very few samples were taken in March or April. In May, a weekly sampling program was established and the samples were taken at that frequency from May to the present. These samples were taken through the normal sample panel known as HPR-227. This sample system only had the capability to take samples from one location in the Reactor Building. The sample point was from the dome area of the Reactor Building, and the piping to the sample panel was several hundred feet long. The exact sampling location is in doubt because a drain valve off the sample line inside the building is thought to be open, therefore, part of the sample comes from the dome area and part of the sample comes from the area just inside the containment sample penetration. Because of this inability to know exactly what location in the Reactor Building was being sampled, Metropolitan Edison decided to establish other sample points. Therefore, a separate sample point, just inside the Reactor Building near the 347' elevation, was established. This second location also used the sample panel of HPR-227. Additionally, two other sample locations were used. The penetration R401, which was used to draw the sump sample, was modified to take an airborne sample just above the water in the basement. Penetration R626 near the operating floor was also used to draw another sample from that area.

From the beginning, Metropolitan Edison had difficulty in getting consistent samples from the Reactor Building. These difficulties were due to long runs of piping inherent in the design of HPR-227, procedural difficulties and analytical difficulties. Eventually, however, the sampling program showed that the major isotope of concern remaining in the Reactor Building after the short half-life radioisotopes had decayed was Krypton 85. Initially, large concentrations of Xenon 133, Xenon 131m, and Iodine 131 were also detected. After several months, however, all these items had decayed away such that essentially the only nuclide above its restricted area MPC was Krypton 85. Selected representative air samples are shown in Table I. Table I also shows the best estimate of currently existing airborne activity in the Reactor Building.

EQUIPMENT HATCH GAMMA SCAN

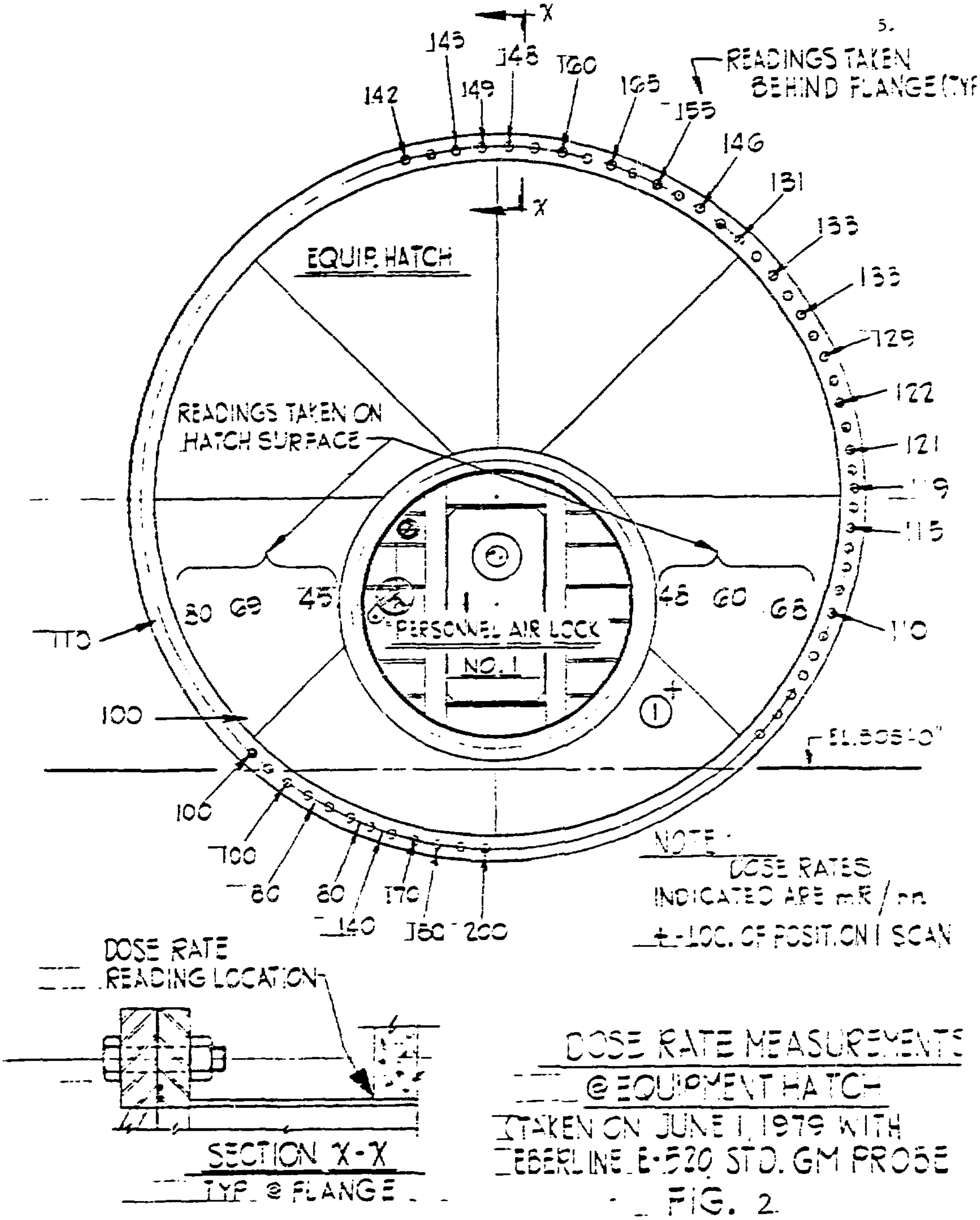
As previously stated, details of this experiment will be given in a subsequent paper by representatives from Bechtel, GPU and Science Applications Inc. Dose readings [2] on the equipment hatch are shown on Figure 2. The major results of this experiment are as follows. The estimated plateout activity on the 305' elevation ranges from 6.3 to 17.3 microcuries per square centimeter. The lower estimate assumes that all of the activity detected in the measurements is from plateout on the vertical surface of the hatch. The upper estimate assumes that the

Nuclide	4/3/79	6/21/79	9/8/79	12/2/79	1/8/80	Current Best Estimate
Kr-85	9.6E-1	7.9E-1	7.8E-1	9.8E-1	1.02E+0	1.04E+0
Xe-131m	—	8.5E-2	—	<MPC	<MPC	<MPC
Xe-133m	5.9E+0	—	—	↓	↓	↓
Xe-133	—	2.4E-2	—	↓	↓	↓
Xe-135	1.6E-1	—	—	↓	↓	↓
I-131	1.1E-1	1.2E-4	—	↓	↓	↓
Cs-134	—	2E-9	3.4E-7	↓	↓	↓
Cs-137	—	9E-9	1.5E-6	↓	↓	↓
Tritium	—	—	—	—	—	4E-5

TMI-2 AIR SAMPLE RESULTS (1)

All Values in μCi/cc

Table I - Ref [1]



DOSE RATE MEASUREMENTS
@ EQUIPMENT HATCH
TAKEN ON JUNE 1, 1979 WITH
EBERLINE E-520 STD. GM PROBE

FIG. 2

Ref [illegible]

activity is based on plateout on the 305' elevation floor. The dose rate on the 305' elevation due to this plateout ranges from 177 to 457 mr/hr. The lower and upper dose rate numbers make the same assumptions as those described for the surface activity numbers above. The major activities found at the 305' elevation are from Cesium and Lanthanum. Iodine 131 was also determined in significant amounts at the time of the measurement, however, essentially all of this Iodine 131 has since decayed.

RADIATION SURVEY THROUGH THE R605 PENETRATION

In order to determine radiation and contamination levels in the basement area of the containment building, measurements were taken in penetration R605 which is approximately 2' above the water level in the containment building. As previously stated, detailed results of this experiment will be presented in a subsequent paper by representatives of Bechtel and Science Applications Inc. This experiment was performed by cutting holes in the outer flange of an existing spare electrical penetration (R605) and inserting a high range gamma survey instrument (teletector) into the penetration. Additionally, a photon spectrum from the water was measured through the penetration using a Ge(Li) detector.

The maximum dose rate measured inside the penetration was 31 R/hr. The 31 R/hr was extrapolated using analytical methods to determine that the dose rate at the surface of the water is approximately 123 R/hr.

From the Ge(Li) readings, it was determine that the major activity contributor in the sump water is Cesium 137 and that it is present in amounts of approximately 366 microcuries per cubic centimeter. The radiation levels [3] measured through penetration R605 are shown in Table II. The estimate of sump inventory [3] resulting from the measurements is shown in Table III.

SUMP WATER SAMPLE

In order to plan and engineer a water cleanup system to treat the water remaining in the sump of the Three Mile Island Reactor Building, a sample was taken from the water. In order to take this sample, the outer flange of penetration R401 was removed and a hole was drilled through the inner flange of that penetration. R401 is located approximately 2 feet above the water. A sample probe was then dropped into the sump water and samples were drawn from the top, middle and bottom of the approximately 7 feet of water existing inside the building. The sump sample was analyzed at Oak Ridge National Laboratory and it was determined that the sump water contains approximately 270 microcuries per millileter activity. The major constituents are Cesium 137, Cesium 134 and Strontium 89/90. The sample from the bottom of the water in the building also showed a greenish precipitate which was determined to be mainly Copper. Table IV shows the radiochemical analyses of the solutions and of the precipitate [5] which was separated from the samples. Table V shows the amounts of Uranium and Plutonium [5] found in each of the samples.

In addition to drawing sump samples through penetration R401, the 4 inch diameter painted steel plug cut from the inner flange of the penetration was removed and sent to Oak Ridge for analysis. Activity present on the plug was found to be mostly Tellurium, with appreciable amounts of Cesium and Niobium also present. Table VI shows the results of the isotopic analysis of activity on the painted steel plug. [5]

Table II - Ref [3]

PENETRATION R605 RADIATION SURVEY RESULTS

POSITION	DOSE RATE (R/hr)
1	27
2	30
3	31
4	16
5	3
6	.800
7	.490
8	.300
9	.220
10	.180
11	.130
12	.100
13	.080
14	.060
15	.085

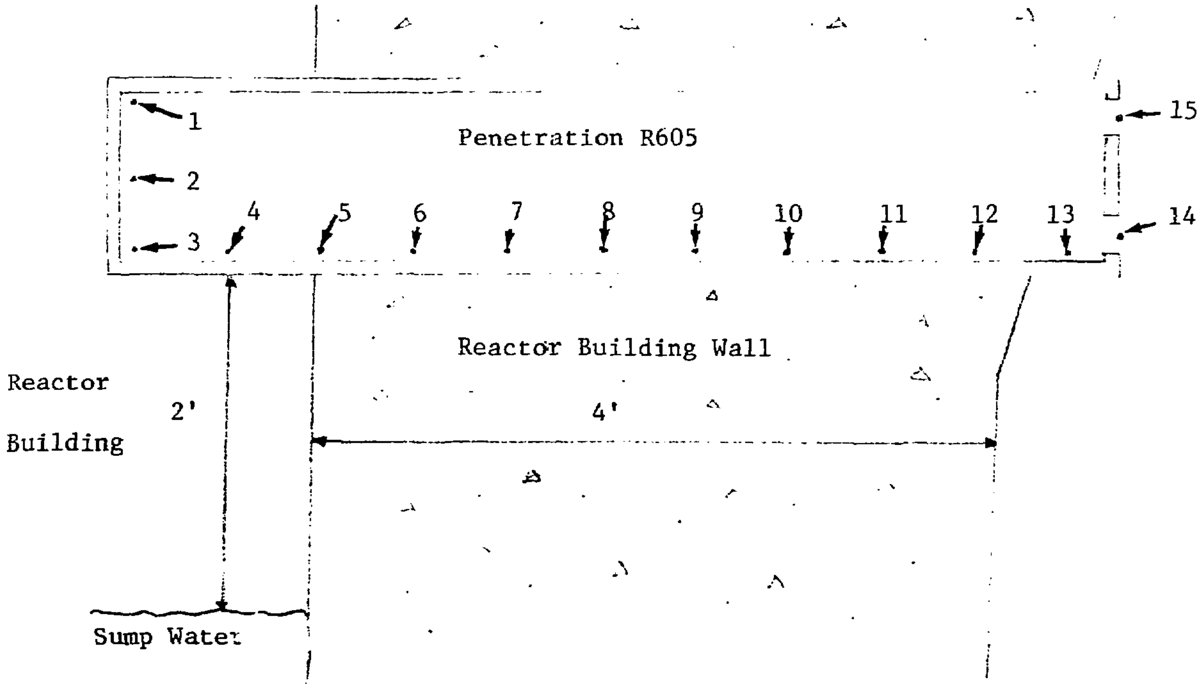


Table III - Ref [4]

PREDICTED SUMP INVENTORY JUNE 20, 1979

ISOTOPE*	ACTIVITIES ($\mu Ci/cm^3$)
XE-131M	7.64E-01
XE-133M	4.74E-15
XE-133	1.09E-10
I-129	5.93E-05
I-131	2.51E+01
I-132	1.74E-05
I-133	1.20E-22
SR-89	2.97E+02
SR-90	9.37E+00
TE-129M	2.40E+00
TE-129	1.54E+00
TE-131M	7.91E-18
TE-131	1.44E-18
TE-132	1.69E-05
BA-137	1.45E+02
BA-140	6.17E+00
RU-103	5.68E+01
RU-106	9.21E+00
LA-140	7.10E+00
CE-141	6.95E+01
CE-143	4.92E-15
CE-144	6.75E+01
PR-143	1.51E+01
PR-144	6.75E+01
EU-155	1.78E-01
EU-156	5.31E-01
ND-147	4.22E+00
NB-95M	2.78E+00
NB-95	1.02E+02
MO/TC99	3.06E-06
Y-89M	2.67E-02
Y-90	9.37E+00
Y-91	2.86E+01
CS-134	4.70E+01
CS-135	5.52E-01
CS-136	1.91E+00
CS-137	1.54E+02
ZR-95	1.28E+02
AG-110	5.77E-02
TOTALS	1.26E+03

*Tritium (H-3) activity in the sump is estimated at 0.5 to 1.5 $\mu Ci/cm^3$ based on normalization to the 6-19-79 RCS sample using Cs-137 as a queing isotope.

Table IV - Ref [5]

RADIOCHEMICAL ANALYSES OF THREE SOLUTIONS

(μCi/ml at 0800, 8/28/79)

Isotope	Top	Middle	Bottom
^{137}Cs	176	179	174
^{134}Cs	40	40	39.6
^{140}La	0.09	0.078	0.14
$^{89+90}$Sr	46.3	43.5	44.9
^{3}H	1.03	1.05	1.01
^{129}I	0.79[a]	0.080[a]	0.076[a]
^{131}I	0.012	0.012	0.013
^{90}Sr	2.70	2.90	2.83
Activity in scavenging precipitation with $Pr(OH)_3$			
^{95}Zr	-----	0.0030	0.0025
^{95}Nb	0.0021	0.0030	0.0099
^{103}Ru	0.005	0.0050	0.0071
^{106}Ru	0.0039	0.0072	0.0099
^{113}Sn*	-----	-----	0.0016
^{125}Sb	0.012	0.015	0.017
^{129}Te	-----	-----	0.035
^{134}Cs	0.0066	0.0059	0.0042
^{137}Cs	0.029	0.028	0.0175
^{141}Ce	-----	0.00047	0.0019
^{144}Ce	-----	0.0046	0.0080
^{140}La	0.036	0.028	0.052
^{140}Ba	-----	0.0038	-----
Gross a	3.4 ± 1.6[b]	1.2 ± 1.3[b]	5.4 ± 2[b]

[a] Units are μg/ml

[b] Units are dpm/ml

*Tentative identification

Table V - Ref [5]

SOLUTION ISOTOPIC ANALYSIS

Sample	Top	Middle	Bottom
U, ppb	7	13	28
234, %	0.021	0.014	0.021
235, %	1.98	1.34	2.04
236, %	0.058	0.036	0.066
Pu, ppb	0.010	0.011	0.033
239, %	89.1	89.4	89.8
240, %	8.5	8.4	8.1
241, %	2.3	2.1	2.0
242, %	-----	-----	Assume 0.1

GAMMA SCAN THROUGH PENETRATION R626

Prior to cutting the inner flange of penetration R626 (in order to insert a camera into the Reactor Building), gamma readings and sodium iodide detector readings were taken through the penetration. As previously stated, details of this experiment will be given in a subsequent paper by representatives from Bechtel and Science Applications, Inc. The major purpose of this experiment was to obtain an estimate of plateout activity on the operating floor 347' elevation.

The gamma survey readings showed maximum dose rates inside the penetration of 50 mr/hr. Table VII shows the dose rates measured in the penetration [6]. Using the information from this Table, it has been calculated that the dose rate at the 347' elevation is approximately 297 mr/hr.

The sodium iodide scan showed mostly Cesium and Barium/Lanthanum gamma peaks. The major energies detected by the sodium iodide detector [6] are shown in Table VIII. Using the energy levels determined by the sodium iodide measurements, estimates of the dose rate and plateout activity on the 347' elevation were made. These are also shown in Table VIII. Cesium 134 was determined to be in the largest concentrations and Cesium 137 was also found plated out in large amounts.

RADIATION MAPPING OF THE NUMBER 2 PERSONNEL AIR LOCK

Since the initial entry into the Reactor Building will be through personnel air lock number 2, experiments were performed to determine the airborne activity, plateout activity and dose rates inside the air lock. The initial experiments consisted of taking an air sample through the air lock vent valve and by inserting radiation probes through a hole provided by removing a pressure gauge from the outside air lock wall. Plateout activity swipe samples will not be performed until an air lock entry is performed.

To perform air activity measurements, an Eberline Ping-2A air monitor was attached to the air lock vent valve. This monitor was used to measure noble gas, iodine and particulate activity in the air lock atmosphere. Additionally, a Marinelli gas sample bottle was inserted in the radiation monitor flow path in order to obtain a direct sample for independent verification of the activity measured by the Eberline monitor. The air lock air sample showed detectable levels of Krypton 85 and Xenon 131M. Krypton 85 in the air lock was found to be 2 x 10^{-3} microcuries per cubic centimeter. Xenon 131M activity was found to be 8 x 10^{-6} microcuries per cubic centimeter. Iodine 131 was found to be present at approximately 1.5 x 10^{-8} microcuries per cubic centimeter. All these activities, i.e. Krypton 85, Xenon 131M and Iodine 131 are above their restricted area MPC. They are, however, several orders of magnitude lower than the activities for these isotopes inside the Reactor Building. The air sample results show that some activity from the Reactor Building has found its way into the air lock by some undetermined mechanism.

Radiation survey readings were taken inside the air lock with a gamma probe and with TLD chips fixed to a probe. The gamma probe was mounted on the end of a 3/8" diameter metal tube which was inserted into the air lock through the hole provided by removing the air lock pressure gauge. The probe was inserted to the inner door. The calcium fluoride TLD chips were similarly mounted on a tube and the tube was inserted to the inner door. The gamma probe readings were made by an Eberline PRM-4 dose rate meter. The readings taken inside the air lock showed that the maximum dose rate was about 100 mr/hr. Table IX shows the results of the readings [7] and Figure 3 shows the points in the air lock at which each reading was taken.

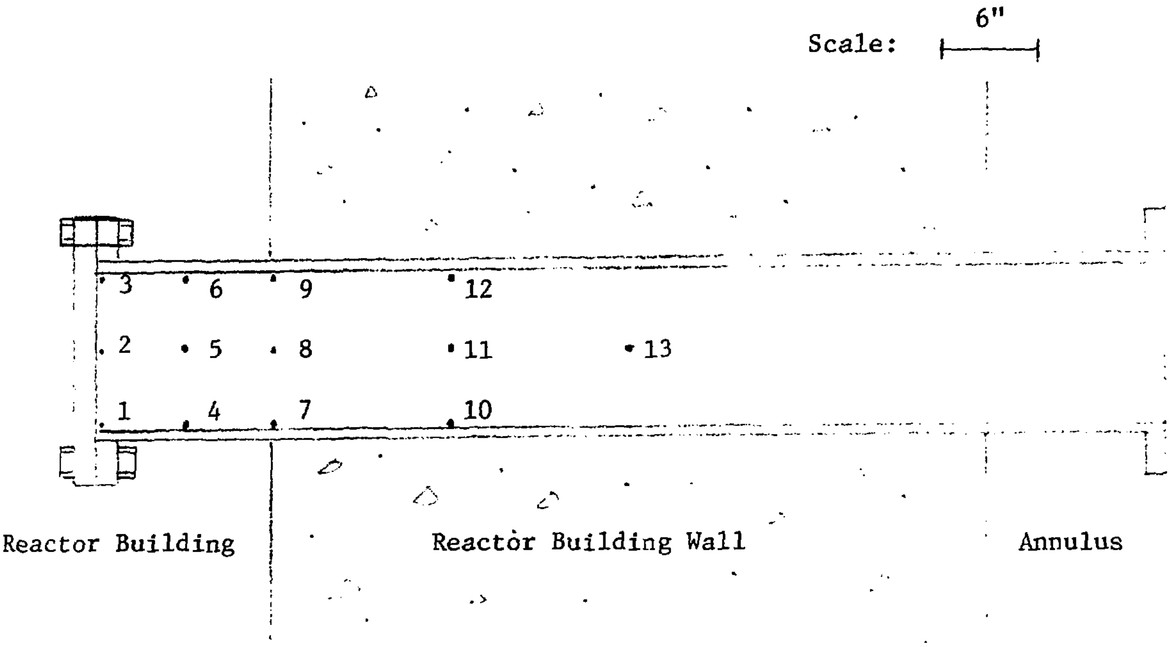
Table VI - Ref [5]

PAINTED STEEL PLUG (μCi TOTAL AT 0800, 8/29/79)

Isotope	μ Ci
^{58}Co	0.032
^{60}Co	0.01
^{95}Zr	0.09
^{95}Nb	1.7
^{103}Ru	0.58
^{106}Ru	0.42
^{110m}Ag	0.080
^{113}Sn	0.24
^{124}Sb	0.005
^{125}Sb	0.45
^{127m}Te	7.8
^{129m}Te	23.6
^{125m}Te	0.5
^{131}I	0.33
^{134}Cs	0.47
^{137}Cs	2.07
^{140}Ba	-----
^{140}La	0.019
^{141}Ce	0.057
^{144}Ce	0.24

Table VII - Ref [6]

PENETRATION R626 GAMMA DOSE RATE SURVEY RESULTS



	Dose Rate (mR/hr)	
Position	Sept. 9, 1979	Oct. 4, 1979
1	30	27
2	35	35
3	35	32
4	40	45
5	50	50
6	45	47
7	15	35
8	25	40
9	15	32
10	1.5	3
11	1	3
12	1	3
13	0.6	-

Dose Rates Measured Using Teletector

9/9/79 - Recorded by John Shoemaker, Frank Nichols (Rad Services), and Ed Walker (Bechtel).

10/4/79 - Recorded by Ed Walker (Bechtel).

Table VIII - Ref [6]

SURFACE ACTIVITY AT ELEVATION 347

E (keV)	Isotope	$D_d i$ (mR/hr)	σ_i (μCi/cm^2)
514	Kr-85	0.008	
563 569 604	Cs-134	0.035	1.53
662	Cs-137	0.139	5.76
796 801	Cs-134	0.071	2.15
1168	Cs-134	0.019	17
1368	Cs-134	0.027	11
1596	Ba/La-140	0.013	0.14

D_d = 0.312 mR/hr

Table IX - Ref [7]

AIRLOCK GAMMA SURVEY DATA

POSITION(d)	DISTANCE FROM FRONT BULKHEAD (ft)	GM/PRM-2 (mR/hr)	CaF2 - TLD(a)(e) (mR/hr) 0°	90°	180°	270°
1	0	30	52.8	40.6	29.2	41.0
2	0.5	60	60.2	71.2	61.0	56.9
3	1.0	80	80.5	(b)	82.1	75.2
4	1.5	60				
5	2.0	-	80.0	-	71.1	-
6	2.5	50				
7	3.0	-	100.9	-	120.5	-
8	3.5	40				
9	4.0	-	60.7	-	74.2	-
10	4.5	20				
11	5.0	-	58.0	-	66.6	-
12	5.5	19				
13	6.0	-	54.1	-	51.6	-
14	6.5	15				
15	7.5	-	84.1	-	(c)	--
16	7.5	9				
17	8.5	8				
18	9.5	9				
19	10.5	8				
20	0.0	60				
21	0.0	50				
22	0.0	40				
23	~1.0	100				
24	~2.5	100				

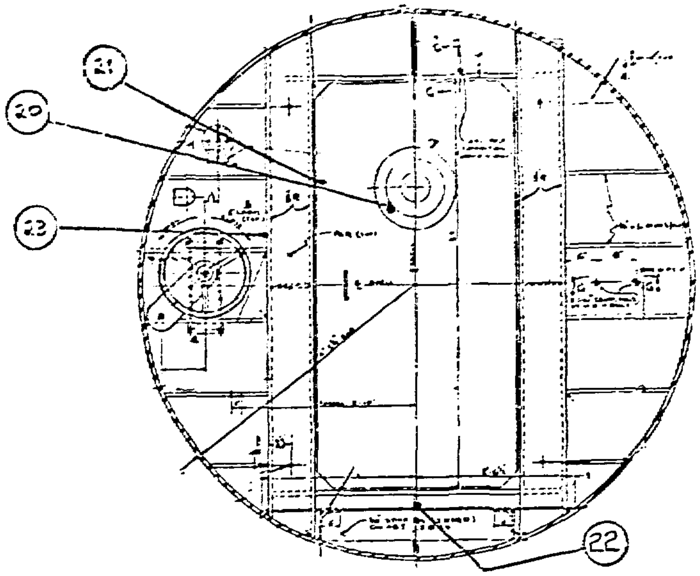
(a) See Figure 2 for the angular orientation of the TLD chips.

(b) TLD chip was broken upon removal from airlock.

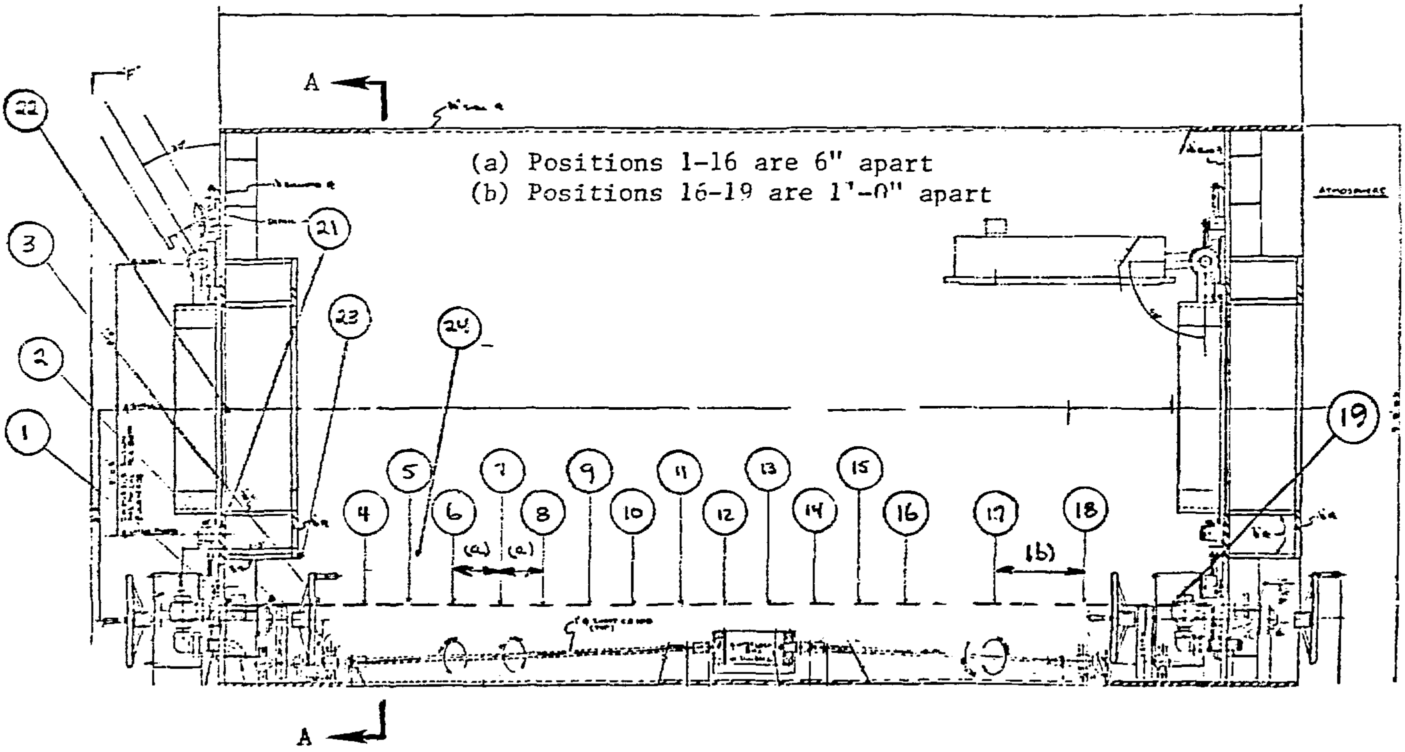
(c) TLD chip was lost upon removal from airlock.

(d) See Figure 3 for location of dose points.

(e) Tip of the TLD rod touching front bulkhead during measurement time.



VIEW A-A



(a) Positions 1-16 are 6" apart
(b) Positions 16-19 are 1'-0" apart

Figure No. 3 Airlock No. 2 Gamma Survey Map

Ref [7]

HYDROGEN RECOMBINER SPOOL PIECE ANALYSIS

Part of the experimental program to determine airborne and plateout activity in the Reactor Building involved the removal of the inlet spool piece to the hydrogen recombiner to determine what isotopes plated out during its operation. The recombiner operated for several days during the first weeks of the accident and as a result, it is thought to contain plateout representative of that which occurs in the early stages of such an accident. Also, the plateout may be representative of that currently in the Reactor Building. The spool piece was removed and shipped to Oak Ridge National Laboratory for analysis. The analysis requested included gamma spectrum measurements, cutting the spool piece into two pieces and then performing beta/gamma spectrum measurements and elemental and compound analyses of the plateout on one side of the pieces. The second piece is stored in TMI archives. Information expected from the results include beta and gamma spectrum results, total number of curies of each isotope detected on the spool piece, elemental, isotopic and compound determination of the plateout, and transuranics that may exist. Results from the analysis performed on the spool piece have not yet been obtained.

REMOTE CAMERA AND RADIATION MEASUREMENTS THROUGH PENETRATION R626

In order to obtain the first direct radiation measurements inside the building and the first remote viewing of the containment, Metropolitan Edison removed the outer flange and cut a hole in the inner flange of penetration R626. The detailed methods and results of this experiment will be presented at the American Nuclear Society 1980 annual meeting in Las Vegas, June 8-13. The title of the paper to be presented at the conference is "Remote Camera/Radiation Monitoring Experiments in the TMI-2 Containment Building."

The initial experiments planned through penetration R626 included camera insertion, radiation monitor insertion, including beta probes and gamma probes, a direct air sample, humidity reading, temperature reading and swipes taken off the Reactor Building wall and off the flange of the penetration. Subsequent experiments included insertion of various other radiation monitors into the penetration and insertion of a frame which had TLD's, film badges and dosimeters mounted at various locations throughout the frame. This frame was also used to determine the beta shielding effectiveness of several types of materials being contemplated for the suit to be worn by initial entry team members.

The camera inspection of the building showed no damage, showed some dust or dirt on the floor, and showed some condensation which resulted in rain in the Reactor Building. Difficulty in obtaining accurate and consistent gamma and beta radiation measurements was experienced. Part of the problem was the interference of Krypton 85 in the operation of the instruments used. For example, the 390 Rad/hr beta reading (Table X) was later shown to be incorrect high, as a direct result of Krypton interference. The range of gamma and beta radiation measurements [8] in the penetration is shown in Table X. Swipes taken in penetration R626 showed mostly Cesium 137 and Cesium 134. Table XI summarizes the results of these swipes [8]. Air samples taken inside penetration R626 confirmed that Krypton 85 was the major isotope present in the Reactor Building air. Further experiments may also be performed as needed through penetration R626 in order to support the initial entry into the Reactor Building.

Table X - Ref [8]

DIRECT RADIATION MEASUREMENT RESULTS (PENETRATION R626)

GAMMA READINGS

TELETECTOR: Gamma Dose Rate = 300 mR/hr

EBERLINE RMS-2: Gamma Dose Rate = 350 mR/hr

SELF-READING DOSIMETER: Gamma Dose Rate = 375-525 mR/hr

FILM BADGE: Gamma Dose Rate = 350-950 mR/hr

TLD: Gamma Dose Rate = 600-925 mR/hr

BETA READINGS

PARALLEL PLATE IC: Beta Dose Rate = 390 Rads/hr

FILM BADGE: Beta Readings = 21-33 Rads/hr

TLD: Beta Readings = 20-44 Rads/hr

BETA DOSE CALCULATIONS: Based on TLD/Film Badge Readings = 100-350 Rads/hr

Battelle Method = 160 Rads/hr
NCRP-44 Method = 205 Rads/hr
NRC Reg. Guide Method = 290 Rads/hr

Table XI - Ref [8]

PLATEOUT SWIPE ISOTOPIC ANALYSIS

Cesium 137	$2 \times 10^{-1} \rightarrow 4 \times 10^{-1}$ µCi/swipe
Cesium 137	$4 \times 10^{-2} \rightarrow 7 \times 10^{-2}$ µCi/swipe
*Strontium 89	$1 \times 10^{-1} \rightarrow 7 \times 10^{-2}$ µCi/swipe
*Strontium 90	$3 \times 10^{-2} \rightarrow 8 \times 10^{-2}$ µCi/swipe
Niobium 95	$9 \times 10^{-4} \rightarrow 3 \times 10^{-3}$ µCi/swipe
**Cobalt 58	$9 \times 10^{-5} \rightarrow 2 \times 10^{-4}$ µCi/swipe
Cobalt 60	$9 \times 10^{-5} \rightarrow 2 \times 10^{-4}$ µCi/swipe

*Not detected on wall or penetration flange swipes.

**Not detected on floor swipes.

AIR LOCK ENTRY

The first step in the actual entry into the Reactor Building will be an entry into the number 2 personnel air lock. This entry consists of opening the outer air lock door while leaving the inner door shut. This entry will allow swipe surveys inside the air lock, Ge(Li) scans through the inner air lock door and viewing of the 305' elevation through the port hole on the inner air lock door. This entry will also afford the opportunity to inspect the outer door seals to determine if deterioration has occurred since the accident. If deterioration is determined to have occurred, these seals can be replaced during this entry.

Prior to this entry, airborne activity in the air lock will be removed by running the sample system hooked up to the air lock vent valve and discharging the activity into the plant ventilation system and through the plant stack.

SUMMARY

The Containment Assessment Program has produced valuable information which was absolutely necessary in order to plan a Reactor Building purge and an initial entry into the Reactor Building. The results of the experiments have shown that the radiological environment inside the building is less hazardous than originally contemplated shortly after the accident. The results show that manned entry into the Reactor Building is feasible and that manned entry can be accomplished with or without Reactor Building purge.

The information obtained from these experiments is also being analyzed in order to allow Bechtel to plan the initial steps in the Reactor Building decontamination. However, detailed planning of this containment decontamination and recovery cannot occur until an initial entry and survey of the Containment Building is accomplished. This building entry should occur in the near future since the information determined by the Containment Assessment Task Force shows that entry can be accomplished by a trained entry team without causing doses for any single individual to exceed the 10CFR20 quarterly allowable doses.

Accomplishment of the experiments initially contemplated has also contributed to an increased level of confidence that the Reactor Building environment can be determined through the conscientious use of a well thought out assessment program. This confidence gained will allow the use of the techniques learned to conduct further experiments as deemed appropriate and necessary to support the Reactor Building recovery. The Containment Assessment Program has been successful and has given an initial step toward the recovery of the Three Mile Island Unit 2 plant.

ACKNOWLEDGEMENTS

The real work of gathering data and conducting experiments was performed by T. C. Menzel (GPU), J. Tate (Gilbert Associates), E. Walker (Bechtel), and T. Fritz (Bechtel).

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DETERMINATION OF THREE MILE ISLAND CONTAINMENT BUILDING WATER LEVEL AND SPECIFIC ACTIVITY OF Cs-137 BY GeLi MEASUREMENTS THROUGH PENETRATION R-605

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ABSTRACT

As part of the TMI-2 Containment Assessment Task Force Planning efforts, the photon spectrum from the sump water was measured at a spare electrical penetration (R-605) using a Ge(Li) detector. The photopeak gamma-ray flux measured at the penetration from Cs-137 was used to estimate the specific activity of Cs-137 in the sump water. The estimate was performed using the three dimensional point kernel gamma-ray shielding code, QAD-CG. The specific activity of Cs-137 in the sump water was determined to be 366 μCi/cm^3. Later Oak Ridge National Laboratory radiochemical assays on samples of the sump water reported concentration of 176 μCi/cm^3. An agreement within about a factor of two.

These gamma scan measurements through penetration R-605 also indicated that the cesium 137 photopeak vanished when the detector was pointed away from the water. Therefore, it was conjectured that the vanishing point inferred a water level of 6.5 feet. This value corresponded to the same level inferred from a barometric level measurement (Heise gauge) of the sump level.

1.0 PURPOSE AND SUMMARY

A Princeton Gamma Tech Ge(Li) detector in conjunction with a Canberra 8180 Analyzer was used to take measurements at electrical penetration R-605. The location of penetration R-605 is shown in Figure 1(3). These radiation measurements of the water in the basement of the containment were used to predict the specific isotopic activity concentration of Cs-137 and to estimate the level of the water in the basement .

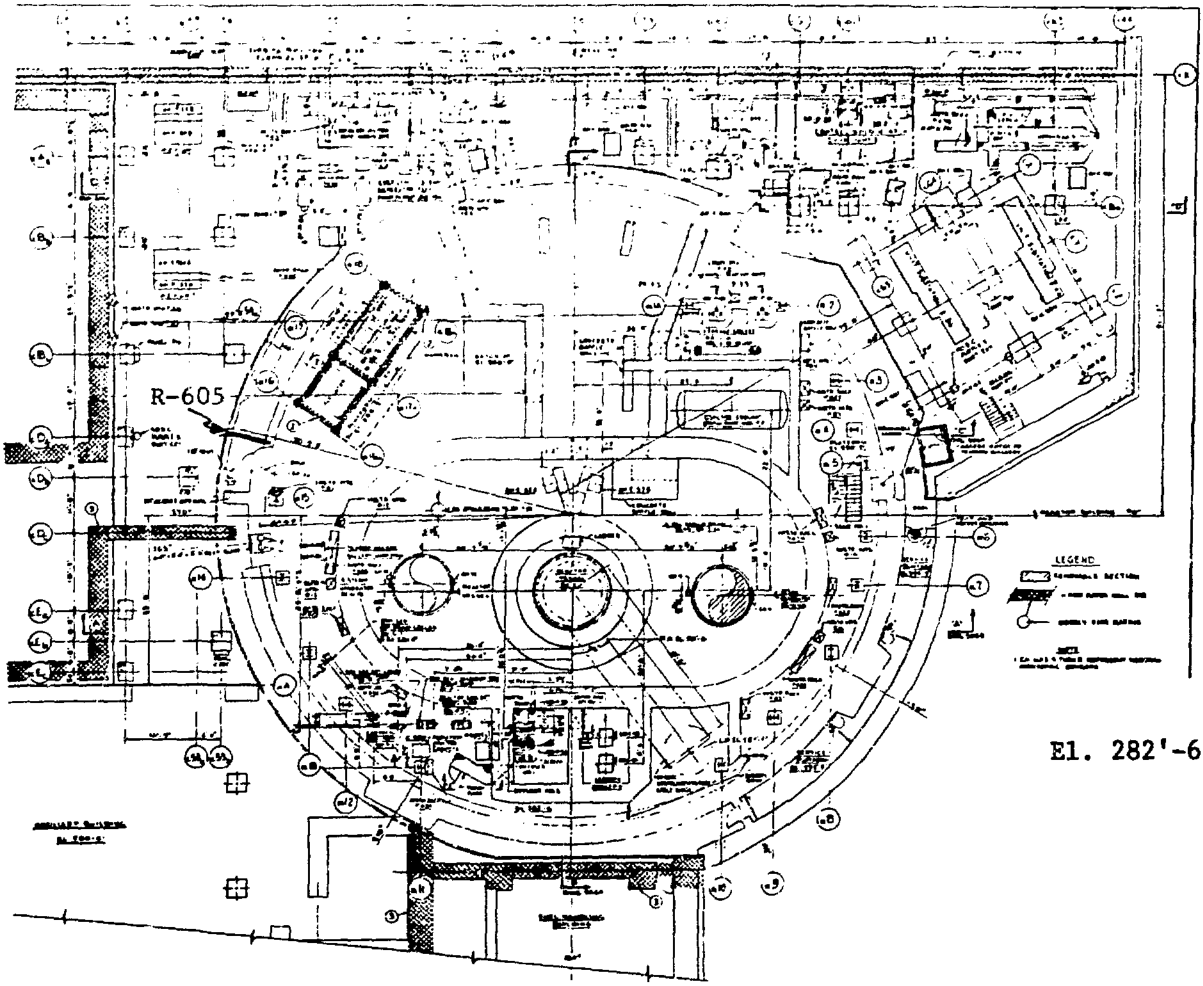

FIGURE 1 LOCATION OF PENETRATION R-605

The first series of measurements consisted of changing the angle of the Ge(Li) collimator with respect to the penetration pipe centerline and recording the counts in the Cs-137 peak. At the angle that the cesium peak vanished, the water level inside the containment was assumed to intersect the steam generator shield wall (see Figure 1).

The estimated water level using this technique was 6 1/2 feet as of 6/22/79, the date of the measurement. This compares to the values measured with a Heise Gauge of approximately 7 feet. Cesium-137 was the only isotope considered for the isotopic analysis of the water since it will dictate waste management planning.

Using the Ge(Li) measurements taken at penetration R-605, the concentration of Cs-137 in the sump water was estimated to be 366μCi/cm^3. This value is in reasonable agreement with the actual sump sample analysis done by Oak Ridge National Laboratory (ORNL). The Cs-137 activity in the sample from the upper region of the sump was 176μCi/cm^3. The theoretical prediction was, therefore, within a factor of 2 of the ORNL analysis.

The radiation measurements and evaluations were performed as a joint effort by the following organizations:

SAI	- J. E. Cline
GPU	- T. C. Menzel
Bechtel	- W. C. Hopkins, E. Walker, D. S. Williams

2.0 REFERENCES

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3.0 WATER LEVEL MEASUREMENTS

3.1 Method

The level of the water is determined as the point at which the detector inside a lead collimator no longer "sees" the cesium-137 activity. This condition is represented as a "line-of-sight" between the collimator and the intersection of the water surface and the steam generator shield wall ("D"-Ring). This geometry is shown schematically on Figure 2. The water level is determined from the relationship:

$$WL = Y - X \tan(\alpha + \beta) \qquad (1)$$

WHERE: Y - vertical height from the containment floor to the centerline of the collimator

X horizontal distance from the "D"-Ring wall to the centerline of the collimator

α - angle of the collimator with the horizontal

β - optical angle of the collimator hole.

$$\beta = \tan^{-1} D_c / L_c \qquad (2)$$

WHERE: D_c - collimator hole diameter = 0.5 cm

L_c - length of collimator hole = 4.0 inches

$\therefore \beta = 2.817^\circ$

The horizontal distance X is determined as the sum of the distance from the "D"-Ring wall to the end of the spare penetration pipe and the distance from the pipe to the collimator centerline. The vertical distance Y is determined as the sum of the distance from the containment building floor to the bottom of the spare penetration pipe and the distance from the pipe to the collimator.

$$X = X_p + X_c \qquad (3)$$

$$Y = Y_p + Y_c \qquad (4)$$

The dimensions relative to the pipe are taken from facility drawings. The dimensions relating the collimator to the pipe were scaled during the experiment. These dimensions are shown on Figure 3.

$$X_p = 29' - 5\ 1/2'' = 353.5 \text{ inches} \qquad (5)$$

$$Y_p = 8' - 11'' = 107.0 \text{ inches} \qquad (6)$$

3.2 Results

Three separate counts were taken for the cesium-137 peak as a function of the collimator angle. The results of these counts are shown in Figure 4. The vertical tilt distance (B_α) is taken from the graph as 1.5 inches .

The corresponding tilt angle (α) is then given by

$$\sin\alpha = \frac{B_\alpha}{A_\alpha} \quad \text{WHERE:} \quad A_\alpha = 24\ 7/8'' \text{ (measured)} \qquad (7)$$

$$\therefore \quad \alpha = 3.457^o$$

Using the geometry in Figure 3, the vertical and horizontal distances from the penetration pipe are determined as follows:

$$X_c = 3\ 3/8 + 7\ 3/4 \cos\alpha - 7\ 3/8 \sin\alpha \qquad (8)$$

$$X_c = 10.666 \text{ inches}$$

$$Y_c = \frac{7\ 3/8}{\cos\alpha} + \sin\alpha\ (7\ 3/4 - 7\ 3/8 \tan\alpha) + 3\ 5/8 \qquad (9)$$

Substitution of these values into the expression for water level yields:

$$WL = (118.454) - (364.166)\tan\ (2.817 + 3.457) \qquad (10)$$

$$WL = 78.42'' = 6.53 \text{ feet}$$

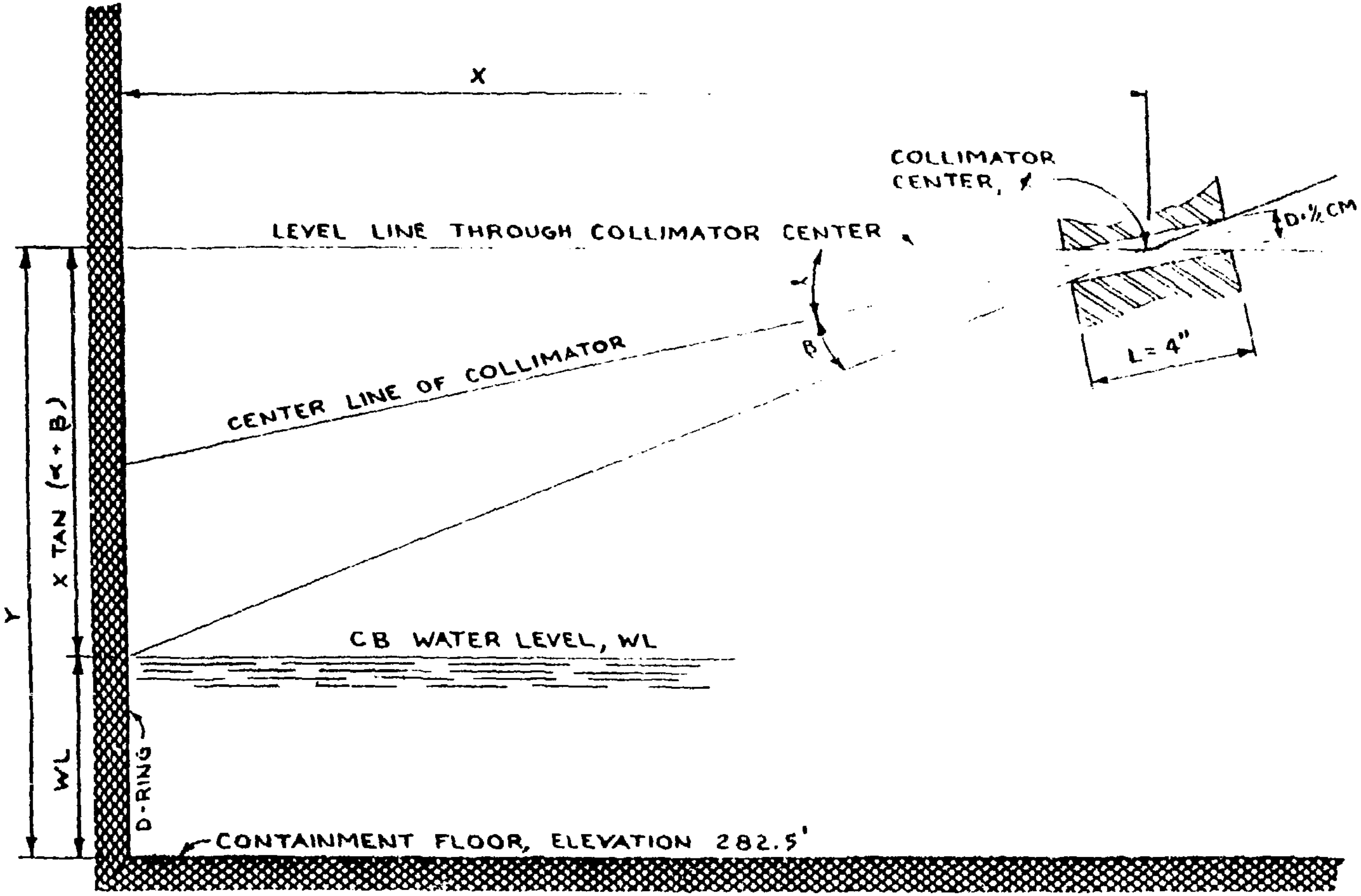
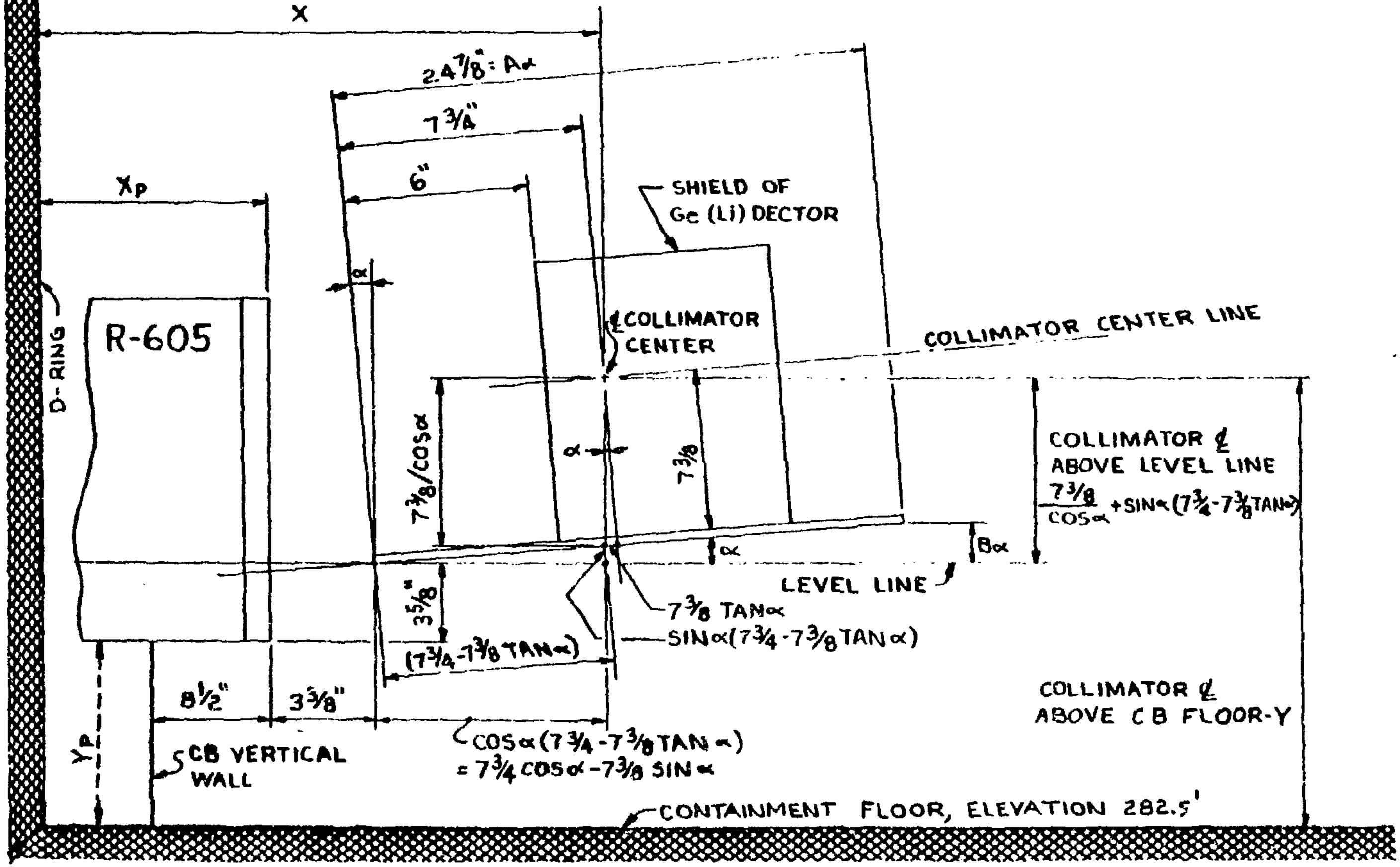


FIG. 2 CALCULATION OF CONTAINMENT WATER LEVEL

FIGURE 3

POSITION OF Ge(Li) DETECTOR WHEN Cs-137 PEAK VANISHES AND CALCULATION OF COLLIMATOR CENTER, ℄, LOCATION, X-Y, WITH RESPECT TO CB FLOOR AND D-RING



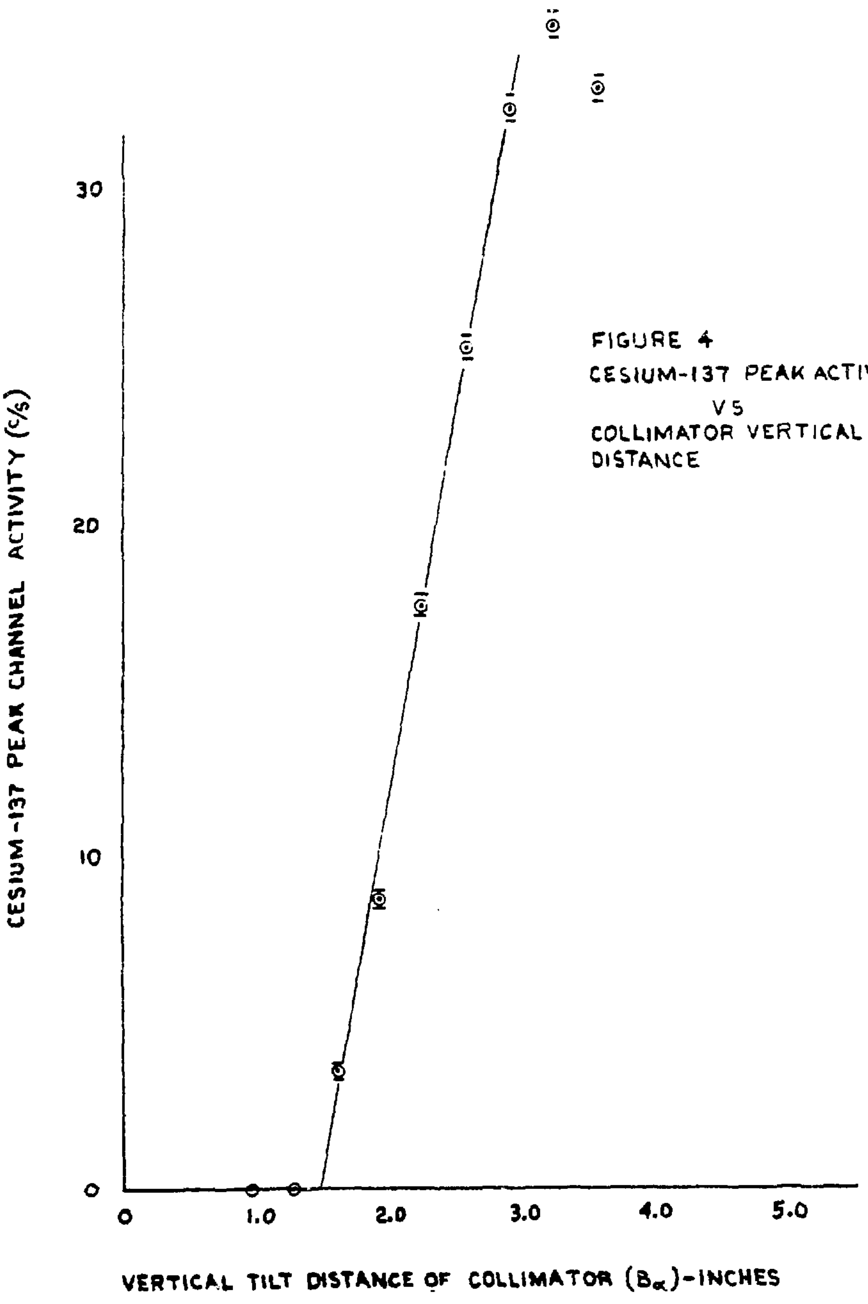
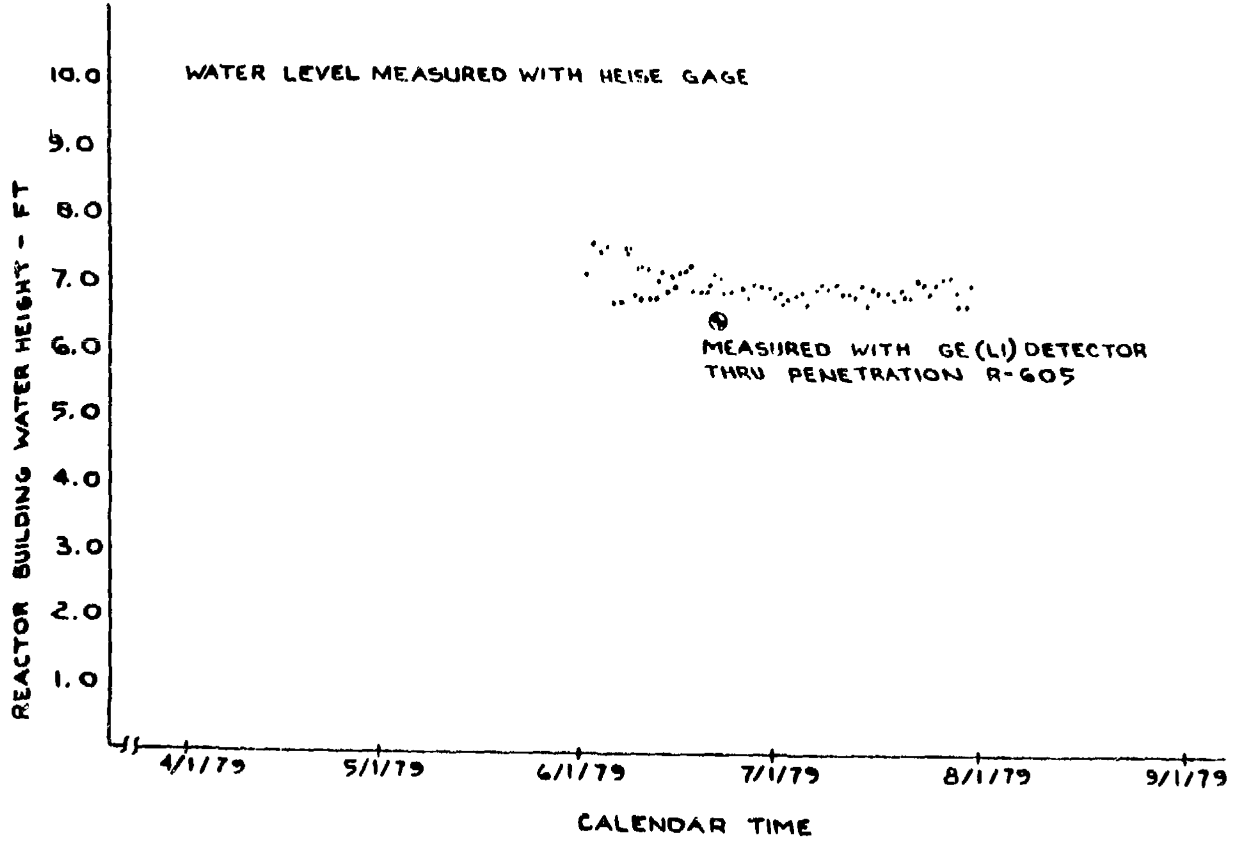


FIGURE 4
CESIUM-137 PEAK ACTIVITY
VS
COLLIMATOR VERTICAL TILT
DISTANCE

FIGURE 5

WATER LEVEL INSIDE TMI UNIT-2 REACTOR BUILDING



This value is compared to the water level measurements recorded daily from the Heise Gauge on Figure 5.

4.0 CESIUM-137 ACTIVITY IN SUMP WATER

4.1 Method

4.1.1 Introduction

The specific activity of the Cs-137 in the sump water was determined by correlating the Ge(Li) gamma-ray photopeak measurements at penetration R-605 (see Figures 8 and 9) with the results of a three dimensional shielding analysis. The shielding analysis was done in an iterative manner with the gamma-ray point kernel code QAD-CG by assuming a unit source in the source volume over which QAD-CG integrates. It was also assumed that the Ge(Li) photopeak flux was entirely due to the Cs-137 in water with no contribution from plateout on the R-605 penetration inboard flange. This assumption was validated during the Ge(Li) measurements by taking a direct collimated scan parallel to the surface of the sump water and normal to the R-605 penetration inboard flange. This scan showed the Cs-137 photopeaks to be predominantly coming from the sump water.

4.1.2 Geometry Model

A three-dimensional model of the containment was assembled using combinatorial geometry (CG). Combinatorial geometry constructs three dimensional configurations using intersections and unions of various geometrical figures such as spheres, cylinders, polyhedrons, etc. The model used included all structures which could attenuate the sump dose rate readings. The containment model included a description of the steel containment liner; three concrete floors, located at elevations 282'-8", 305', and 347'-6"; steam generator compartments ("D" rings); penetrations 605 and 626; Ge(Li) detector collimator; personnel locks 1 and 2; the equipment hatch; and the sump water. In addition to checking the dimensions used in the geometry model, a visual check of the model was made by taking two dimensional slices of the model and comparing them with design drawings. The two dimensional slices were made using an auxiliary program called PICTURE.

Figures 6 and 7 are two views of the full containment model. Details near penetration R-605 are given in Figures 8 and 9. Note that the sump water being viewed by the Ge(Li) detector is also shown in these figures. The detector collimator is shown separately in Figures 10 and 11, but does not appear in earlier figures because the resolution used in the PICTURE slices is too gross to pick up the collimator.

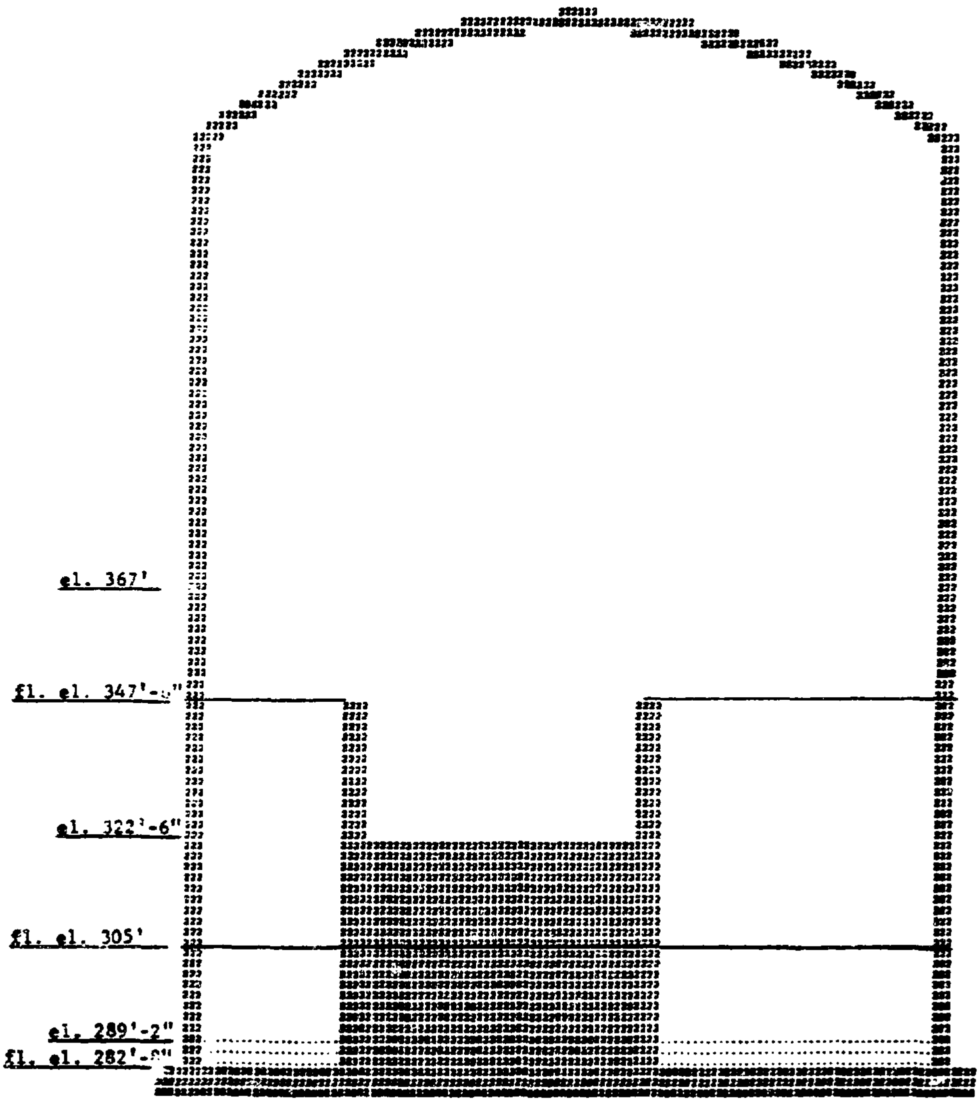
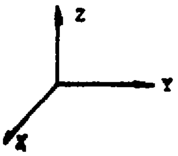


FIGURE 6 FULL CONTAINMENT, X=0

KEY:
2=concrete
3=steel
·=water



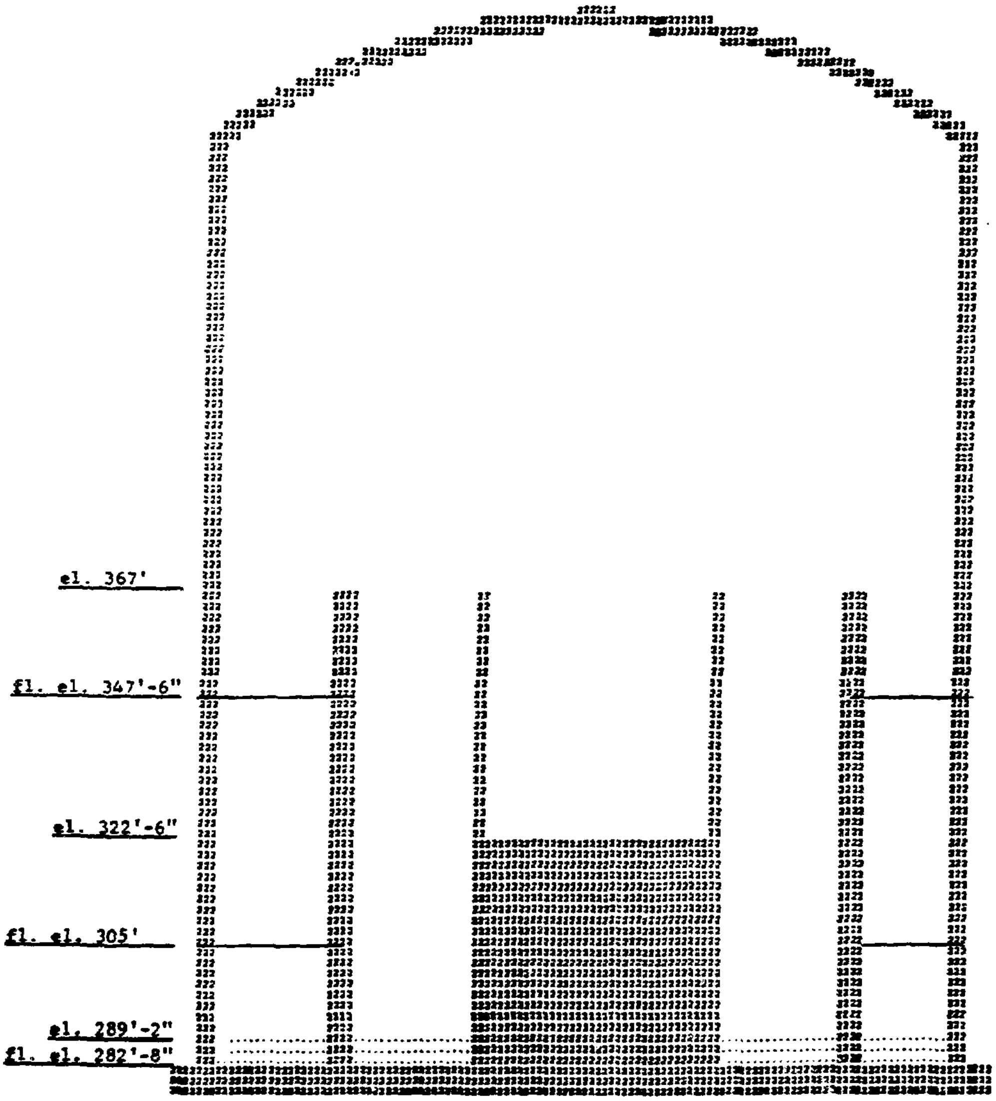
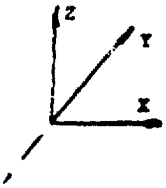


FIGURE 7 FULL CONTAINMENT, Y=0



KEY:
2=concrete
3=steel
·=water

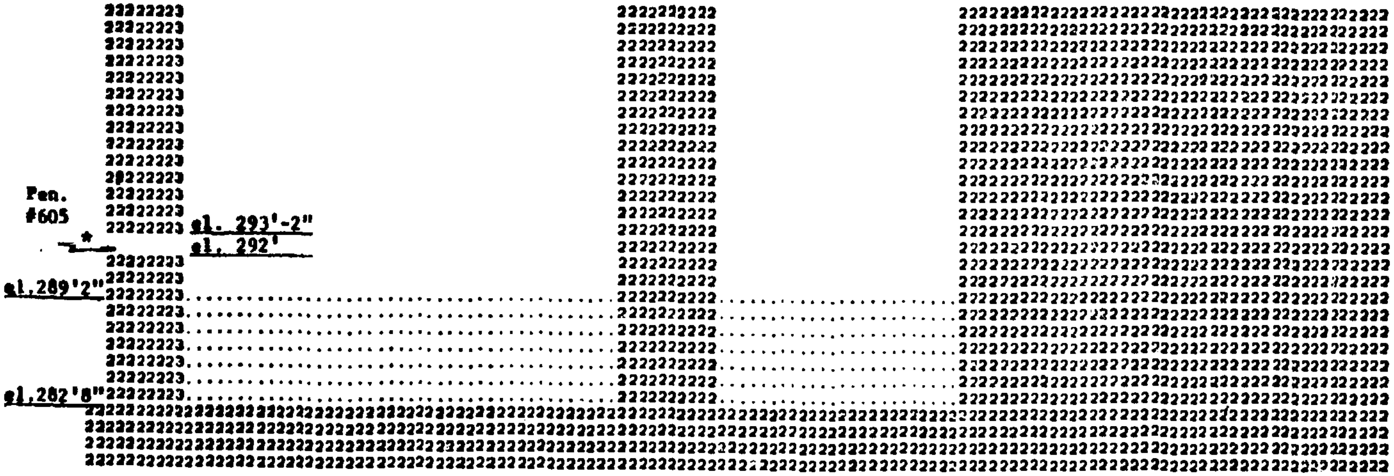


FIGURE 8 PENETRATION #605 WITH SUMP WATER AND D. RING

KEY:
2=concrete
·=water
3=steel
*=detector location

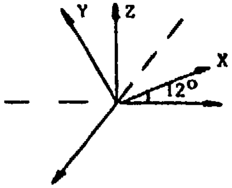


FIGURE 9 PENETRATION #605 WITH SUMP WATER

KEY:
2=concrete
3=steel
.= water
*=detector location

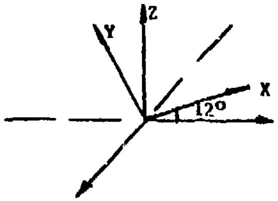


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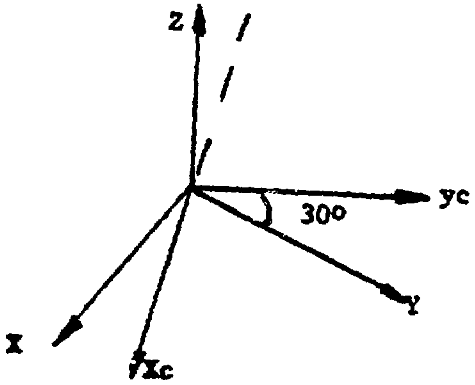
FIGURE 10 COLLIMATOR

KEY:
1=lead



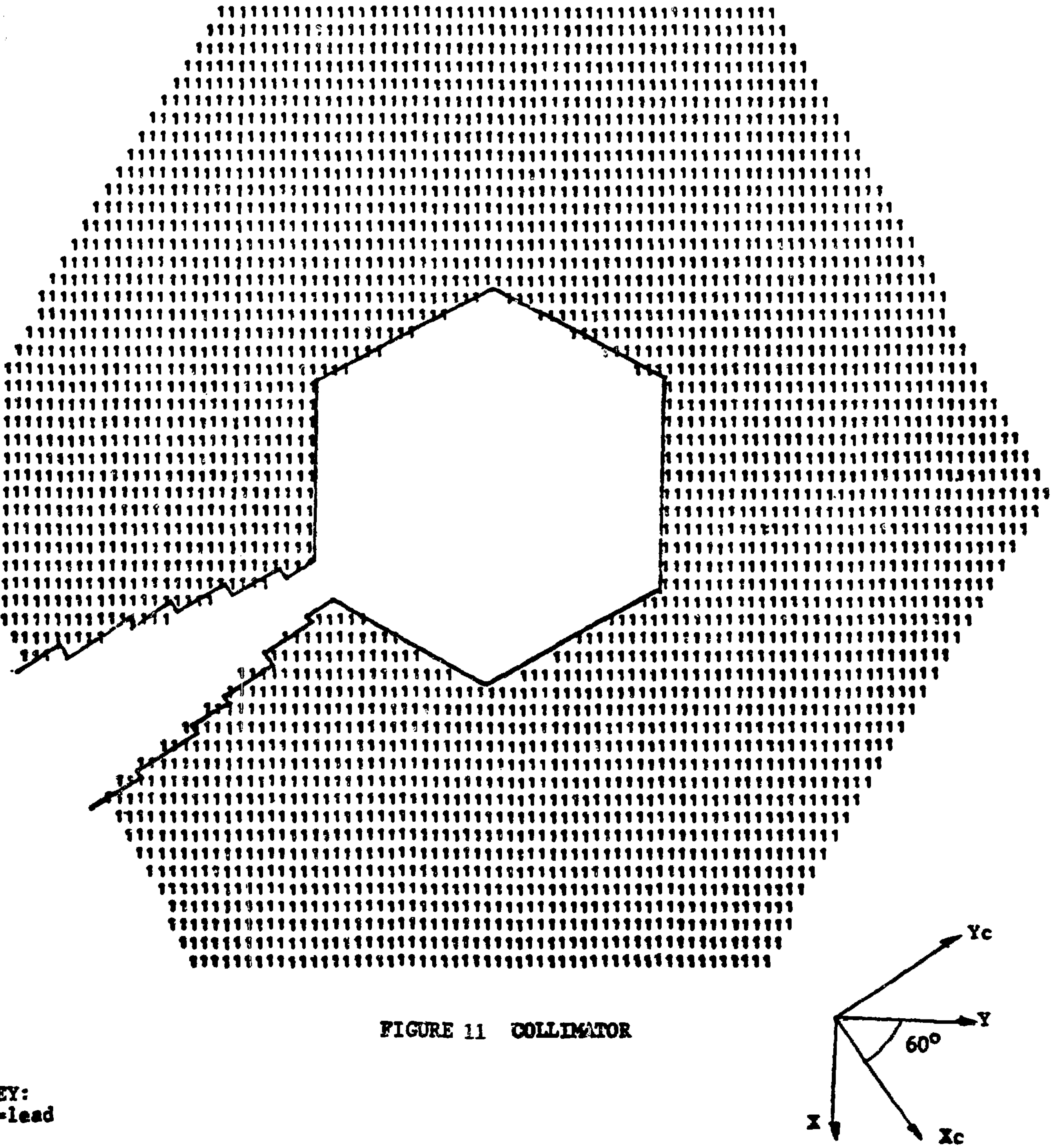


FIGURE 11 COLLIMATOR

EY:
=lead

4.1.3 QAD-CG Analysis

QAD-CG$^{(2)}$ uses a point kernel ray tracing technique. The point kernel represents the contribution to a receptor point along a "line-of-sight" path from a point isotropic source. This is then combined with a buildup factor which accounts for the contribution at the receptor point due to scattered gamma rays. Since the source is actually distributed over a volume, the point kernel must be integrated over the source volume. The source volume is nodalized, forming a collection of isotropic point sources. The magnitude of these point sources is equal to the product of the node volume and the volumetric source strength at the node point. The total dose rate at a receptor resulting from a point isotropic source is expressed as:

$$D(\vec{r}) = K \int_V \frac{S(\vec{r}\,')\, B(\mu|\vec{r} - \vec{r}\,'|, E)\, \exp(-\mu|\vec{r} - \vec{r}\,'|)}{4\pi|\vec{r} - \vec{r}\,'|^2}\, dV \quad , \qquad (11)$$

where

$\vec{r}$ = point at which gamma dose rate is to be calculated,

$\vec{r}\,'$ = location of source in volume V,

V = volume of source region,

μ = total attenuation coefficient at energy E,

$|\vec{r} - \vec{r}\,'|$ = distance between source point and point at which gamma intensity is to be calculated,

$B(\mu|\vec{r} - \vec{r}\,'|, E)$ = dose buildup factor,

K = conversion factor (flux-to-dose rate) at energy E.

The buildup factors used in QAD-CG are based on the data by Goldstein and Wilkins$^{(12)}$ for gamma-ray transport in an infinite homogeneous source for a point isotropic source. Capo's third degree polynomial expression$^{(11)}$ of the Goldstein and Wilkins data is used for the QAD-CG buildup factor algorithm.

The intent of this was to estimate the specific activity level of Cs-137 in the sump water. Unlike most QAD-CG shielding analyses, it was necessary to rigorously define the source volume. This is because most shielding analyses are done in a forward (as opposed to adjoint) mode with a given volumetric source strength. In such analyses, the shielding engineer is concerned only with assuring that adequate conservatism in the shield design provides a prudent safety margin. For the case now being analyzed, a different situation arises because this conservatism must now be avoided in order to match theory and experiment. Because QAD-CG calculates only in a "forward" mode, it was necessary to determine in an iterative manner the size of the volume which was most important (in the true transport adjoint sense) to the uncollided flux at the R-605 penetration. Once the optimum source volume was determined, a sensitivity study on the extent of nodalization of that volume was also run to see the effect on the uncollided flux estimate at the R-605 penetration.

This was accomplished by making a series of QAD-CG runs. The source volume description was varied separately in each dimension until convergence was achieved for that dimension. The variable used for determining convergence was the uncollided flux calculated by QAD-CG at the R-605 penetration detector point with the restriction that for each new iteration the source strength per unit volume be held constant throughout all iterations, i.e.,

$$\frac{\int_v S(\bar{r},E)\, dv}{\int_v dv} = \text{constant for all iterations} \qquad (12)$$

Proceeding iteratively in this manner, the uncollided flux followed the trend shown below in Figure 12 as expected.

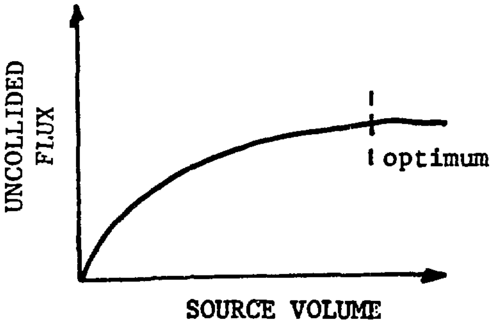


Figure 12, Trend of Uncollided Flux Value as a Function of Source Volume Activity Held Constant.

After the optimum volume (properly nodalized) was determined, the photopeak flux measured at the R-605 penetration for Cs-137 was folded into the normalized QAD-CG result as follows:

Given that:

$$S_{Total} = \int_v S_v \, dV \qquad (13)$$

where S_v is the volumetric source strength in gammas/cm^3, then the following proportion will hold:

$$\frac{\phi m}{S_{Total}} = \frac{\phi'}{S'} \qquad (14)$$

where S' = the total source strength determined by the QAD-CG to be present in the optimum volume.

ϕ' = the QAD-CG calculated uncollided flux at penetration R-605 from the final iteration for the properly nodalized optimum volume.

ϕ_m = the measured photopeak flux for Cs-137

Rewritting equation 14 in terms of specific activity:

$$\frac{\phi m}{S_v} = \frac{\phi'}{S_v'} \qquad (15)$$

where $S_v = \frac{S_{Total}}{V}$ (V is the optimum volume) (16)

and $S_v' = \frac{S'}{V}$; (17)

solving equation 15 for S_v yields:

$$S_v = \frac{\phi m \; S_v'}{\phi'} \qquad (18)$$

substituting for S_v' gives:

$$S_v = \frac{\phi m}{\phi'} \frac{S'}{V} \qquad (19)$$

4.2 Results

Once the optimum volume was determined, a final QAD-CG run was made. This run used the maximum degree of nodalization allowed in the code. The predicted concentration of Cs-137 in the sump water is:

from equation 19

$$S_v = \frac{(4800\ \gamma/\text{cm}^2\ \text{sec})(1.0\gamma/\text{sec})}{(8.69\text{x}10^{-10}\ \gamma/\text{cm}^2\ \text{sec})(4.076\text{x}10^5\ \text{cm}^3)} = 1.354\text{x}10^7\ \frac{\gamma}{\text{sec-cm}^3}$$

or converting units,

$$S_v = \frac{(1.354\text{x}10^7\ \frac{\gamma}{\text{sec-cm}^3})}{(3.7\text{x}10^4\ \frac{\gamma/\text{sec}}{\mu\text{Ci}})}$$

$$S_v = 366\ \mu\text{Ci/cm}^3 \text{ of Cs-137}$$

This value is approximately twice the value predicted in the initial planning study (9).

Note: The iterations started with a unit total source (S' = 1) the final S' ≠ 1 because of the constraint on the volumetric source strength of equation 2.

4.3 Conclusion

The experiment demonstrates the capabilities of the estimated Ge(Li) detector as a versatile technique in making radiological assessments. The water level determination shows reasonable agreement with the Heise Gauge measurements as shown on Figure 5.

The result of the isotopic analysis (360 μCi/cm^3) is in good agreement with the initial prediction (154 μCi/cm^3) in reference 9 for the concentration of Cs-137 in the sump water and the actual sump sample analysis by ORNL (176 μCi/cm^3)(10). The difference between measurement and analytical correlation may be attributed to uncertainties associated with the detector calibration and photon flux determination, the assumptions used to analyze the source photon flux distribution, and the complexity of the source-detector geometry relationships. This type of analysis can also be used to predict the concentrations of other isotopes in the sump water.

Table I(1)

FLUXES OBSERVED THROUGH THE R605 PENETRATION
($\gamma/cm^2/sec$)

Energy (keV)	Run 4
131-I	
364.5	51.
636.0	7.
134-Cs	
563.2	74.
569.3	161.
604.7	1214.
795.8	1454.
801.8	88.
1038.4	24.
1167.8	54.
1365.2	135.
136-Cs	
340.6	10.
818.5	5.
1048.1	24.
1235.3	27.
140-La	
328.8	35.
487.0	114.
537.4	29.
752.0	13.
815.9	3.
868.0	24.
919.6	9.
925.2	26.
1596.5	1228.
2348.1	17.
2521.7	82.
2547.5	2.
137-Cs	
661.6	4809.
85-Kr	
514	

GAMMA-RAY MEASUREMENTS IN CONTAINMENT PENETRATION R-626 AT THREE MILE ISLAND, UNIT 2

J. E. Cline
C. D. Thomas, Jr.
Science Applications, Inc.

and

E. Walker
S. R. Blazo, Jr.
Bechtel Power Corporation

ABSTRACT

On September 9, 1979, a high range gamma survey meter with extendable probe (Teletector) was used to perform a radiation survey inside spare penetration R-626. This penetration is located at elevation 358', approximately 11 feet above the operating floor of the reactor building and is approximately one foot inside the reactor building wall. The results of this survey were reconfirmed with another survey conducted on October 4, 1979. The maximum dose rate observed inside the penetration for either survey was 50 mR/hr.

Following the September 9, 1979, gamma survey, a gamma photopeak spectrum was obtained by inserting into the penetration a lead shielding collimator with a 1.0 cm hole to a 2" x 2" Na I (Tl) detector designed by SAI.(1) This spectrum indicates Cesium 134 and 137 as the principle source of photons with lesser amount of Kr-85 (air activity) and Barium/Lanthanum-140 photopeaks also indicated.

The measured dose rates and gamma spectrum form the basis for estimates of the dose rate at the operating deck (elevation 347 ft.) and the isotopic composition of floor deposition gamma emitters.

The deposition activity was estimated to be as follows:

Cs-134 : $= 1.53\ \mu\text{Ci/cm}^2$
Cs-137 : $= 5.76\ \mu\text{Ci/cm}^2$
Ba/La-140: $= 0.14\ \mu\text{Ci/cm}^2$

The corresponding dose rate at the floor from this activity is

$D_f = 300$ mR/hr

1.0 Purpose and Summary

A high range, extendable probe, gamma survey meter (Teletector) was first used to measure the gamma dose rates inside Penetration R-626. This survey was a measure of the radioactivity from gamma emitting isotopes inside the reactor building as attenuated through approximately 1.0 inches of the steel penetration pipe wall. A collimated gamma spectrum was then obtained inside the penetration pipe using a 2" X 2" NaI (Tl) detector with the collimator hole directed toward the operating floor, approximately 11 feet below the penetration. The attenuated dose rate and the gamma spectrum were combined to estimate the amount of floor deposition and corresponding floor dose rate.

The maximum dose rate measured inside the penetration was 50 mR/hr. The collimated gamma spectrum, reduced to seven primary energy groups, indicated the presence of Cs-134 and Cs-137 with lesser amounts of Ba/La-140 and airborne Kr-85. The floor deposition activity, based on these measurements, is 1.53 μCi/cm^2 of Cs-134 and 5.76 μCi/cm^2 of Cs-137. The corresponding dose rate from the floor source is 300 mR/hr.

The results presented here will be incorporated with future measurements planned for inside the reactor building to more completely characterize the radiation source terms that will be encountered during the decontamination and recovery operations.

The radiation measurements and evaluations were performed as a joint effort by the following organizations:

Rad Services, Inc. - Frank Nichols, John Shoemaker
SAI - James Cline, C. D. Thomas
GPU - Tom Menzel
Bechtel - W. C. Hopkins, S. R. Blazo, Ed Walker

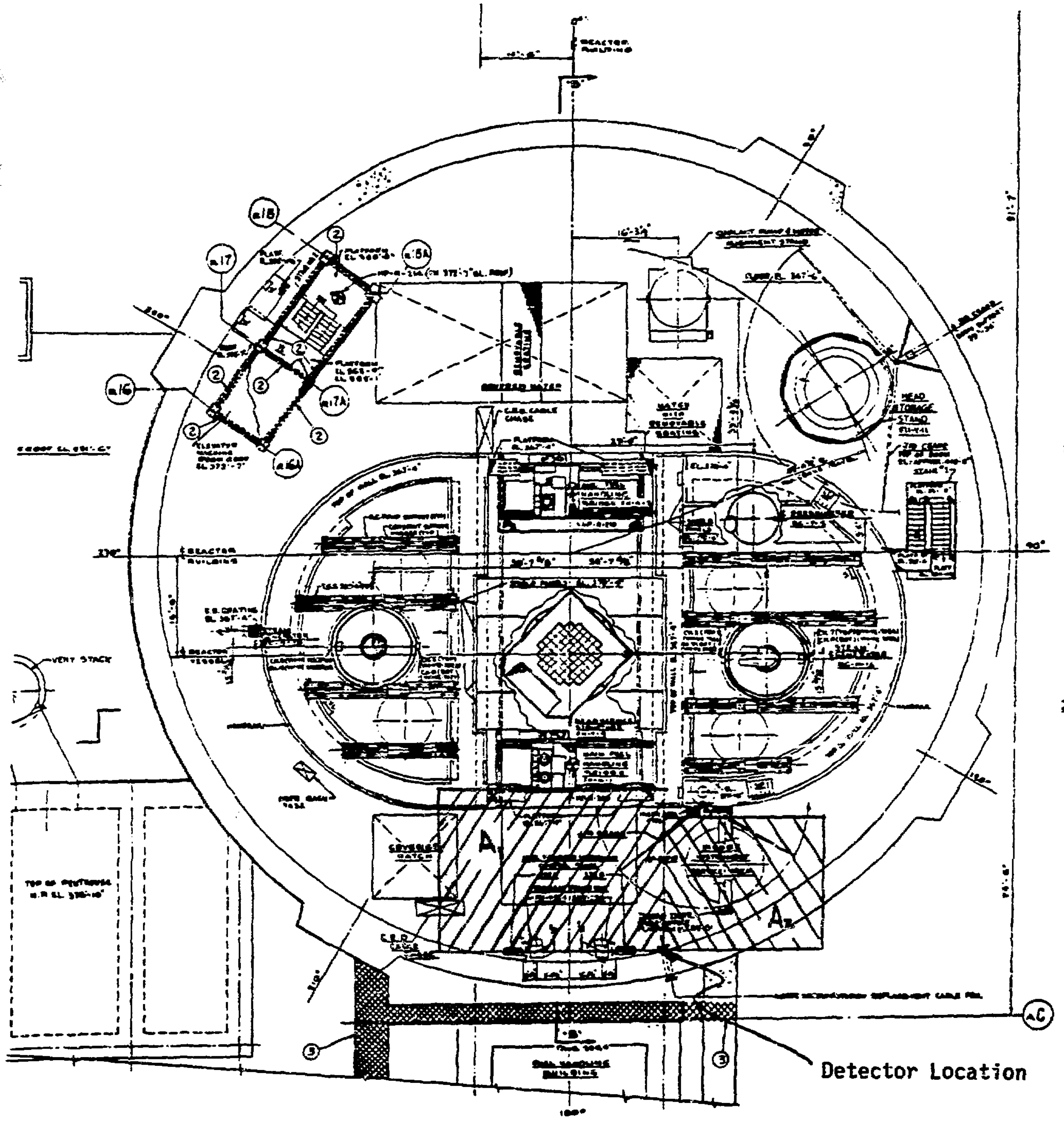


FIGURE 1: PENETRATION R-626 GAMMA DEPOSITION SOURCE GEOMETRY

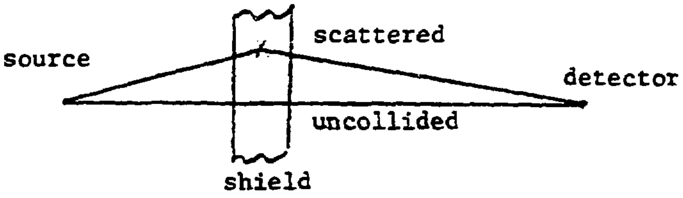
The factor of 2 accounts for the bi - directional (zero net current) assumption for a planar source.

The collimation factor (K_c) is defined as the relationship between the "uncollided" dose rate outside the collimator shielding (Du) and the effective dose rate at the detector (D_d). The detector dose rate is determined from the photopeak fluxes measured by the detector as:

$$D_d = \sum_i K_i \phi di \tag{4}$$

where: K_i = dose rate conversion for photon i (mR/hr/γ/ cm^2-s).

The "uncollided" dose rate outside the detector is found from the measured dose rate inside the penetration pipe (Dm). This dose rate is comprised of photons that have transmitted through the pipe wall directly to the detector point (uncollided) and photons that have been scattered by the pipe wall from a direction away from the detector back to the detector point.



The relationship between the uncollided and total photon flux at the detector point is given by

$$B\ (\mu t) = \frac{Dm}{Du} \tag{5}$$

The collimation factor (Kc) is defined as the ratio of the uncollided dose rates across the collimator, thus:

$$Kc = \frac{Dm}{B(\mu t)\ D_d} \tag{6}$$

2.0 Methods

The deposition activity on the floor and the corresponding dose rate are estimated from the measured dose rate and photopeak spectrum inside the penetration pipe. The methodology used for this evaluation is discussed in the description of the equipment hatch experiments.(5)

The dose rate at the floor is determined for the source-detector geometry relationship shown on Figure 1 using the method of Hubbell(4) For this evaluation the effective plane source is described as two equivalent rectangles (A_1 and A_2 on Figure 1). The relationship between the measured dose rate (D_m) and the corresponding dose rate at the floor (D_f) thus becomes

$$D_m = D_f\ (f_1 + f_2) \cdot B(\mu t) \cdot e^{-\mu t} \qquad (1)$$

where: μ - attenuation coefficient for shield between source and detector
t - shield thickness
$B(\mu t)$ - shielding buildup factor
f_1, f_2 - source/detector geometry factors for source area 1 and 2

$$f = (\tfrac{1}{2}) \tan^{-1} \left[\frac{ab}{\sqrt{1 + a^2 + b^2}} \right] \qquad (2)$$

$a = {}^{L}/z$
$b = {}^{W}/z$

L - length of the equivalent source
W - width of the equivalent source
Z - height of detector above the equivalent source

The isotopic surface source activity on the floor is evaluated from the uncollided photon fluxes measured by the collimated 2" x 2" NaI (Tl) detector. The relationship between the surface activity on the floor and the uncollided photon flux measured by the detector inside the penetration pipe is given by:

$$\sigma_i = \frac{2\ \phi_{di}\ (2.7 \times 10^{-5})\ K_c\ e^{-\mu t}}{\xi_i\ (f_1 + f_2)} \qquad (3)$$

where: σ_1 - surface activity of isotope i (μCi/cm^2)

ϕ_{di} - uncollided photon flux at detector from isotope i (γ/cm^2 -s)

ξ_i - photon yield for isotope i (γ/dis)

K_c - Collimation factor

$2.7\text{x}10^{-5}$ - constant of proportionality (s-μCi/dis)

3.0 Measurements

3.1 Gamma Survey

The gamma dose rates were measured inside the Penetration R-626 pipe on September 9, 1979, using a high range gamma survey meter with extendable probe (Teletector). A similar survey was conducted on October 4, 1979, using another teletector to verify the original results.

The centerline of this penetration is at elevation 358'-6". The survey point at the inside surface of the penetration flange is approximately one foot from the inside of the reactor building wall and eleven feet from the floor of the operating deck (el. 347'-6"). The results of these surveys are shown on Figure 2.

3.2 Gamma Photopeak Spectrum

The gamma photopeak spectrum was obtained by inserting a 3-inch lead shielded 2"x2" NaI (Tl) detector with a 1.0 cm collimator hole into the penetration pipe. The configuration of the collimator and detector is shown on Figure 3. The detector/collimater was positioned against the inner flange of the penetration with the collimator hole directed toward the floor. The location of the collimator hole was approximately four inches from the inner flange (~8 inches from the reactor building wall). The resulting gamma photopeak spectrum is shown on Figure 4 and the corresponding photon fluxes are summarized in Table 1.(1)

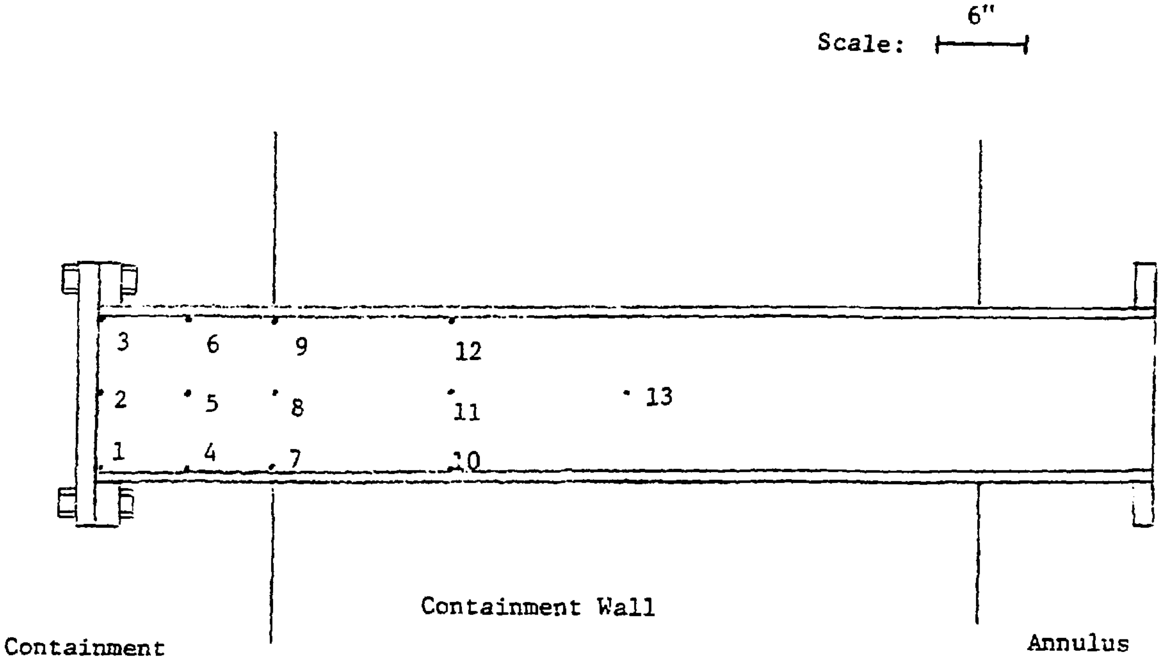
Table 1

Results From The Analysis Of The NaI(Tl) Spectrum Taken On

TMI-2 CONTAINMENT PENETRATION R-626

Gamma-Ray Peak Energy (keV)	Nuclide	Peak Counting (c/s)	Gamma-Ray Flux ($\gamma/cm^2/s$)
1596	^{140}Ba/La	1.22	4.7± 0.2
1368	^{134}Cs	3.14	11 ± 4
1168	^{134}Cs	3.0	9 ± 3
796(+801)	^{134}Cs	16.2	42.3± 0.6
662	^{137}Cs	41.8	96.2± 1.0
604(+563+569)	^{134}Cs	12.3	27.3± 1.0
514(+511)	^{85}Kr(+^{140}Ba/La)	3.5	7.1± 1.0

FIGURE 2: PENETRATION R-626 GAMMA DOSE RATE SURVEY RESULTS



Dose Rate (mR/hr)

Position	Sept. 9, 1979	Oct. 4, 1979
1	30	27
2	35	35
3	35	32
4	40	45
5	50	50
6	45	47
7	15	35
8	25	40
9	15	32
10	1.5	3
11	1	3
12	1	3
13	0.6	-

Dose Rates Measured Using Teletector

9/9/79 - Recorded by John Shoemaker, Frank Nichols (Rad Services), and Ed Walker (Bechtel)

10/4/79 - Recorded by Ed Walker (Bechtel)

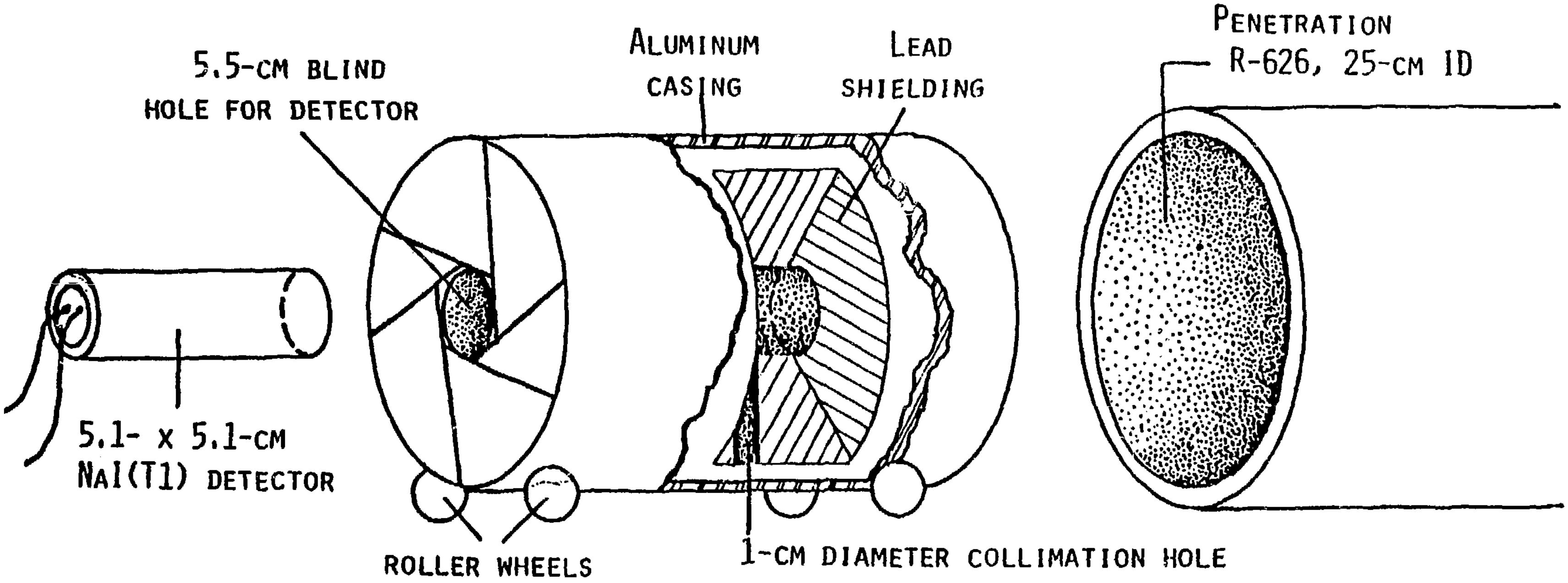


FIGURE 3. SHIELDING FOR AND COLLIMATION 5.1-CM BY 5.1-CM NAI(T1) DETECTOR, SHOWING A CUT-AWAY VIEW OF THE ASSEMBLY.

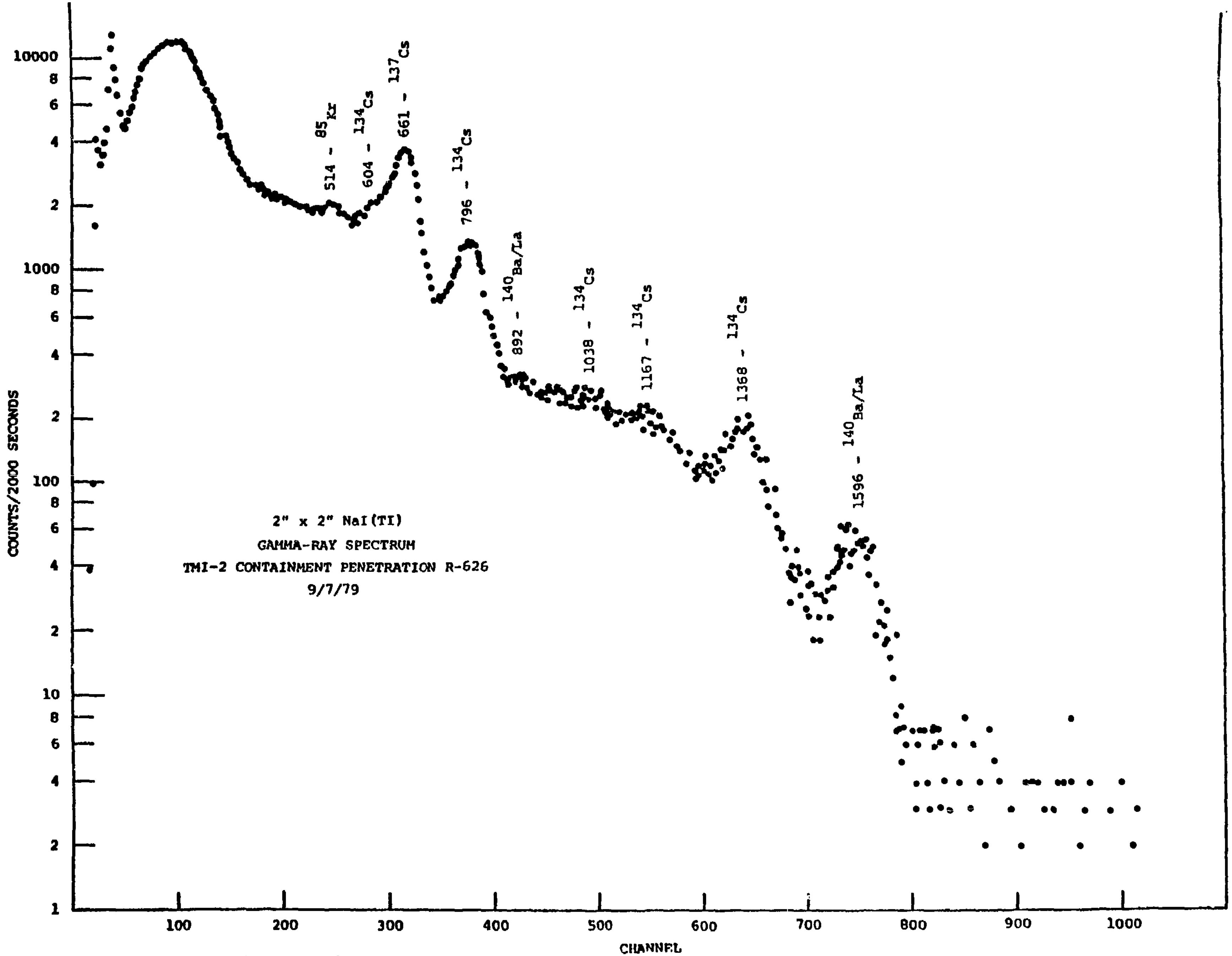


FIGURE 4. GAMMA-RAY PULSE-HEIGHT SPECTRUM FROM TMI, UNIT 2, CONTAINMENT PENETRATION R-626.

4.0 Results

4.1 Dose Rate at the 347'-6" Elevation

The dose rate at the floor of the operating level inside the reactor building was determined from the dose rate measurement inside the penetration pipe assuming that the measured dose rate resulted entirely from radioisotopic deposition on the floor. (Table 2 shows that the Kr-85 activity outside the penetration pipe contributes ~2.5% of the total dose rate inside the penetration pipe). The source geometry factors (f) were evaluated using equation (2) for the following parameters (See Figure 1).

$L_1 = 35$ ft , $L_2 = 25$ ft

$W_1 = 25$ ft , $W_2 = 21$ ft

$Z = 11$ ft , $Z = 11$ ft

$f = 0.169$, $f_2 = 0.151$

The shielding parameters for the 1-inch steel penetration pipe wall are:

$t = 1.0$ inch $= 2.54$ cm

$\mu = 0.55 \text{ cm}^{-1}$ (2) for $\bar{E}_\gamma = 0.7$ MeV

$B(\mu t) = 2.2$ (3)

For the measured dose rate inside the penetration pipe, $D_m = 50$ mR/hr, the dose rate at the floor (D_f) is found by substituting into equation (1) to obtain

$$D_f = \frac{(50) \text{ mR/hr}}{(0.169+0.151)\ (2.2)} \quad e^{(.55)(2.54)}$$

$$D_f = 297 \text{ mR/hr}$$

4.2 Isotopic Deposition at the 347'-6" Elevation

The isotopic distribution for deposition activity at elevation 347'-6" was determined for the assumption that the observed dose rate inside the penetration resulted from all isotopes except Kr-85 located on the floor. (6)

The collimation factor (Kc) for the detector is determined by first calculating the detector dose (D_d) using the photopeak data in Table 1 and solving equation (4). The detector dose rate evaluation is summarized in Table 2 and the total detector dose rate given as:

$$D_d = 0.312 \text{ mR/hr}$$

Table 2

NaI (Tl) Detector Collimated Dose Rate

E_γ (keV)	Isotope	$K_i^{(7)}$ $\left(\frac{mR/hr}{\gamma/cm^2-s}\right)$	$\phi_{di}^{(1)}$ (γ/cm^2-s)	D_{di} (mR/hr)
514	Kr-85	1.17E-3	7.1$\pm$ 1.0	0.008
563 569 604	Cs-134	1.29E-3 1.29E-3 1.36E-3	27.3$\pm$ 1.0	0.035
662	Cs-137	1.44E-3	96.2$\pm$ 1.0	0.139
796 801	Cs-134	1.68E-3	42.3$\pm$ .6	0.071
1168	Cs-134	2.15E-3	9$\pm$3	0.019
1368	Cs-134	2.45E-3	11$\pm$4	0.027
1596	Ba/La-140	2.75E-3	4.7$\pm$ .2	0.013
			D_d =	0.312 mR/hr

For the effective photon spectrum in Figure 4, an effective buildup factor B(µt) was determined$^{(8)}$ using the computer code QAD-CG.$^{(9)}$ This factor is:

$$B(\mu t) = 2.89$$

Substituting these values into equation (6) results in the collimation factor of

$$Kc \simeq 55.8$$

Equation (3) is now solved for surface activity using the detector photon fluxes in Table 1 and the source/detector geometry factors (f) from Section 5.1. This evaluation is summarized in Table 3.

TABLE 3
Surface Activity at Elevation 347

E_γ (keV)	Isotope	ξ_i (γ/dis)	μt(6)	ϕ_{oi} (γ/cm^2-s)	(f_1+f_2)	ϕ_{fi} (γ/cm^2-s)	σ_i (μCi/cm^2)
514	Kr-85	.0041		(Not a plateout source)			
563 569 604	Cs-134	.084 .154 .976	1.759	8845	.320	2.7600	1.53
662	Cs-137	.851	1.688	29033	.320	90700	5.76
796 801	Cs-134	.854 .087	1.530	10901	.320	34100	2.15
1168	Cs-134	.018	1.283	1812	.320	5660	17
1368	Cs-134	.0304	1.182	1991	.320	6220	11
1596	Ba/La-140	.956	1.086	777	.320	2430	0.14

WHERE:

$$\phi_{oi} = \phi_{di} K_c e^{-\mu t} \qquad \phi_{fi} = \phi_{oi}/(f_1+f_2)$$

5.0 Conclusions

Note that the range of equivalent surface source activity in Table 3 for the isotope Cs-134 ranges from 1.53 - 17 μCi/cm^2. This spread may be attributed to the accuracy of the photopeak determination and incomplete collimation of the higher energy photons. Table 4 shows a comparison of the cesium activity results determined in this report and the floor activity predicted at elevation 305'.(5) The best estimate of the Cs-134 activity has been taken as that based upon the lowest energy peaks due to combined abundance (gives better counting statistical accuracy) and more complete collimation of the uncollided flux.

Table 4

Cesium Deposition Activity Comparison

	305' Elevation(5)	347' Elevation
Cs-134	1.13 μCi/cm^2	1.53 μCi/cm^2
Cs-137	4.12 μCi/cm^2	5.76 μCi/cm^2
Deposition Ratio	3.65	3.76
Fission Product Ratio(10)	3.23 (Decayed 6/1/79)	3.46 (Decayed 8/29/79)

6.0 References

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EQUIPMENT MATCH PLATEOUT AND ACTIVITY MEASUREMENTS AT THREE MILE ISLAND*

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ABSTRACT

Prior to decontamination and personnel entry into the TMI Unit 2 containment, the radiation sources must be defined. As part of this effort, a series of dose rate and photon spectrum measurements were made outside the equipment hatch. These measurements were analyzed to provide a preliminary estimate of the magnitude and isotopic distribution of plateout activity on the 305 elevation. The upper and lower bound estimates of this activity are:

(a) Lower bound - plateout on vertical surface of hatch

$\sigma_h = 6.3\ \mu Ci/cm^2$ (surface activity)

$D_h = 177$ mR/hr (dose rate)

(b) Upper bound - plateout on 305 elevation floor

$\sigma_f = 17.3\ \mu Ci/cm^2$ (surface activity)

$D_f = 457$ mR/hr (dose rate)

These estimates should be considered order of magnitude. Further reduction of the collimated data is required to determine both the magnitude of each isotopic contributor and its location as plateout. It should be noted that these dose rates are not those values expected inside containment at elevation 305 as they do not include contributions from activity on the 347 elevation or from the water activity on the containment floor - elevation 282. These dose rates correspond only to the plateout source at elevation 305.

The results of this experiment provide isotopic identification of the plateout activity at elevation 305. The data are summarized for the lower and upper bound scenarios. These data will provide bases for cleanup requirements, re-entry, etc. This information, combined with that of future experiments, will be used to evaluate the total area dose rates expected upon re-entry.

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*Because of the length of the paper that was submitted, it was not possible to publish it in its entirety. Persons interested in obtaining a copy of the full paper should contact the authors.

SESSION XX

LOCA TRANSIENT ANALYSIS

Chairmen

K. J. Brinkmann - Netherlands Energy Research Foundation

R. T. Fernandez - Yankee Atomic Electric Company/Electric Power Research Institute

Dup

LOCA Analyses for Nuclear Steam Supply Systems with Upper Head Injection

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ABSTRACT

The term "Upper Head Injection" describes a relatively new addition to a nuclear reactor's emergency cooling system. With this feature, water is delivered directly to the top of the reactor vessel during a loss-of-coolant accident, in addition to the later injection of coolant into the primary operating loops. Established computer programs, with various modifications to models for heat transfer and two-phase flow, were used to analyze a transient following a large break in one of the main coolant loops of a reactor equipped with upper head injection. The flow and heat transfer modifications combined to yield fuel cladding temperatures during blowdown which were as much as 440K (800°F) lower than were obtained with standard versions of the codes (for "best estimate" calculations). The calculations also showed the need for more uniformity of application of heat transfer models in the computer programs employed.

INTRODUCTION

Analysis of transient thermal-hydraulic phenomena in a nuclear reactor system presents complex computational problems. An integrated, detailed description of all components and physical processes in a nuclear steam supply system is clearly beyond current capabilities, so that various geometrical and physical simplifications are necessary. An important question, therefore, is the way in which such simplifications influence the results obtained through their use.

Sandia National Laboratories has been involved in programs directed toward understanding reactor behavior during a loss-of-coolant accident (LOCA) [1,2,3]. The objective of the LOCA Analysis program is to provide the NRC with an ability to perform independent analyses of pressurized water reactor behavior under postulated accident conditions. One of the principal areas of interest has been the operation of a relatively new form of emergency core cooling system, which employs upper head injection (UHI). UHI is a safety feature which delivers water to the top of the reactor vessel during a LOCA, prior to the injection of emergency coolant into the primary operating loops (see Figure 1). UHI was developed for use with small, ice condenser containment systems. The nature of UHI requires adequate treatment of such phenomena as top-down quenching of fuel rods, separated flow of steam-water mixtures, and fluid transfer from the injection point to the rest of the system.

In this paper we discuss results of analyses performed on the response of a UHI-equipped reactor system following the complete rupture of a primary cooling system pipe. With the large outflow of fluid through the break, the system depressurizes rapidly, and heat transfer between the fuel rods and coolant is diminished. This first portion of the accident, called the blowdown phase, is followed by a period (reflood) during which the emergency cooling systems overcome the rate of loss of fluid, resulting in the return of liquid to the core.

ANALYTICAL METHODS AND RESULTS

The analytical methods used in the studies described here were versions of the RELAP, [4] FRAP, [5] and TOODEE2 [6] computer codes. RELAP4/MOD5, in modes appropriate to both the blowdown and reflood phases, was used to obtain a quasi-one-dimensional approximation to the thermal-hydraulic behavior during the LOCA. Fuel rod-to-coolant heat transfer was only modelled in RELAP during the blowdown phase; during reflood a "carryover-rate fraction" model [7] and constant exit enthalpy were employed, because the standard reflood heat transfer in RELAP was not totally adequate for our purposes. FRAP provided detailed fuel rod conditions for the blowdown phase of the analysis, using coolant boundary conditions from the RELAP results. Fuel rod and rod cladding temperature histories during reflood were obtained using TOODEE2, again using RELAP results for core conditions. The calculated peak clad temperature (PCT) is the quantity of primary importance.

The current RELAP nodalization for the UHI blowdown analysis, which was evolved during the course of the investigation, is shown in Figure 2. Of particular note are volumes 43 and 44, modelling the guide tubes; volume 45, which models the support columns, and the inclusion of the UHI system (volumes 51 and 52). Phenomenological models have been proposed which are specifically intended for safety evaluation analyses of UHI-equipped plants [8]. These models include descriptions of separated two-phase flow with slip, core quenching during blowdown, and heat and mass transfer during reflood. In order to implement these models in our analyses, it was necessary to modify the standard, or generic, versions of the codes mentioned above. By comparing the results of calculations with both the standard and modified models for heat and mass transfer, we can identify some of the features of the models which have a dominant influence on PCT. The complicated nature of the problems being addressed and the strong interactions between various models frequently make the process of identification difficult. In addition, the incorporation of the new slip model in RELAP introduced calculational difficulties during the blowdown phase.

The new model for core quenching specifies the surface heat transfer coefficient for a fuel assembly, independently of the standard heat transfer logic. This specification is in force for a short period of time after various criteria are satisfied in a region adjacent to the assembly. Because of the experimental base used in developing the model, one of these criteria requires the presence of co-current downflow next to the rod. The flow regime is, of course, very sensitive to the technique used to model two-phase flow. Our calculations display significantly different patterns of core quenching during blowdown, depending upon whether or not the new models are used. Figure 3 shows surface temperatures for a fuel rod assembly in the hot bundle; large temperature differences become apparent after UHI flow ceases. In other comparisons, the use of the quench model alone does not produce such large differences in PCT. Thus, as may be expected, the simultaneous implementation of new and interdependent models for complicated phenomena can produce strong synergistic effects in the results.

In RELAP's generic treatment of two-phase flow, a simple correlation depending on void fraction is used to calculate the relative velocity, or slip, between the phases. After the removal of an ambiguity in the definition of void fraction used in the numerical scheme, the generic method yielded relatively efficient and stable calculations. The new model replaces the simple slip correlation with one based on a drift flux model, and depends on a number of parameters in addition to the void fraction [9]. This model produces highly oscillatory flow results with consequentially more time-consuming calculations. In addition, quench patterns produced in the core using the new models appear to be less realistic than those observed without the modified slip correlation and special quench model. Figure 4 shows quenched regions in the generic and modified calculations, at 80 s into the blowdown phase of the transient. In the modified results, it seems peculiar that the entire hot bundle should be quenched while lower power regions in the average core are unquenched. This situation persisted from no later than 25 s to the time shown.

It became evident during the course of our work that critical heat flux correlations for licensing calculations were implemented differently in all the codes used [3]. In the generic calculations, these differences produced clad surface temperature differences as large as 170K (300°F). In the modified results, the special quench model minimized the temperature differences.

Full licensing calculations were performed with modified and generic versions of the codes. Peak clad temperatures for the entire transient were found to be 1340K (1950°F) and 1300K (1890°F) respectively. Our results showed that a licensing heat transfer criterion had a much larger effect on peak clad temperature than did any of the variations on thermal-hydraulic models. This criterion forced the hot pin to unquench (in FRAP calculations), regardless of local coolant and clad surface temperature conditions.

The results of our calculations provide some information on the way in which models for hydraulic and thermal phenomena can interact to produce strikingly different results in reactor system analysis. Furthermore, the calculations show that consistency in treating models for various phenomena should be considered when combining results of different computer codes to analyze a complete LOCA.

ACKNOWLEDGEMENTS

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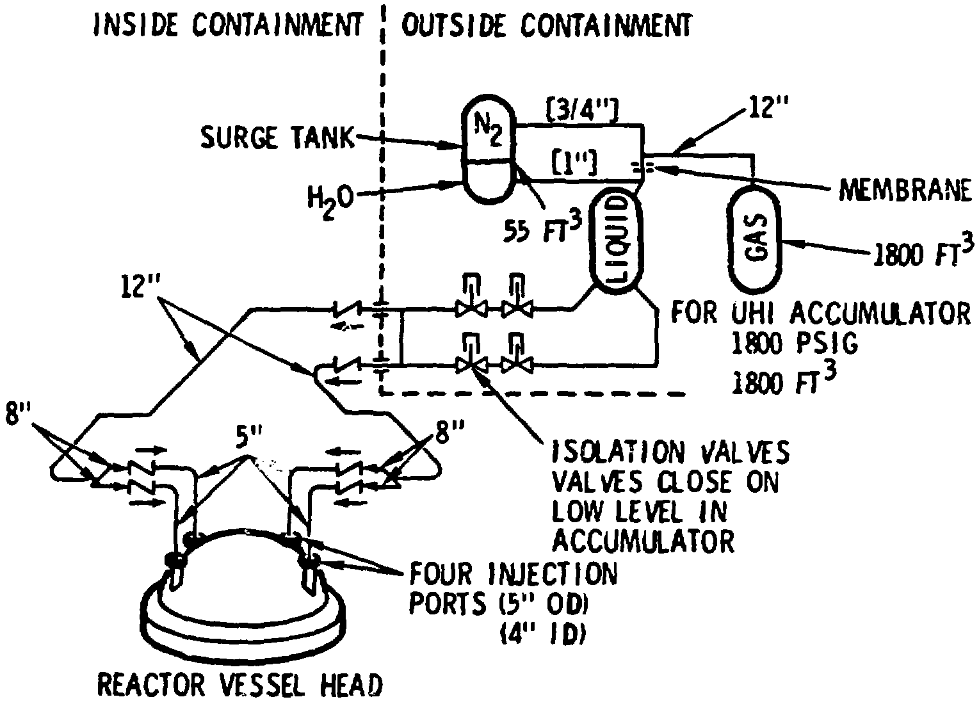


Figure 1. Upper Head Injection System Schematic

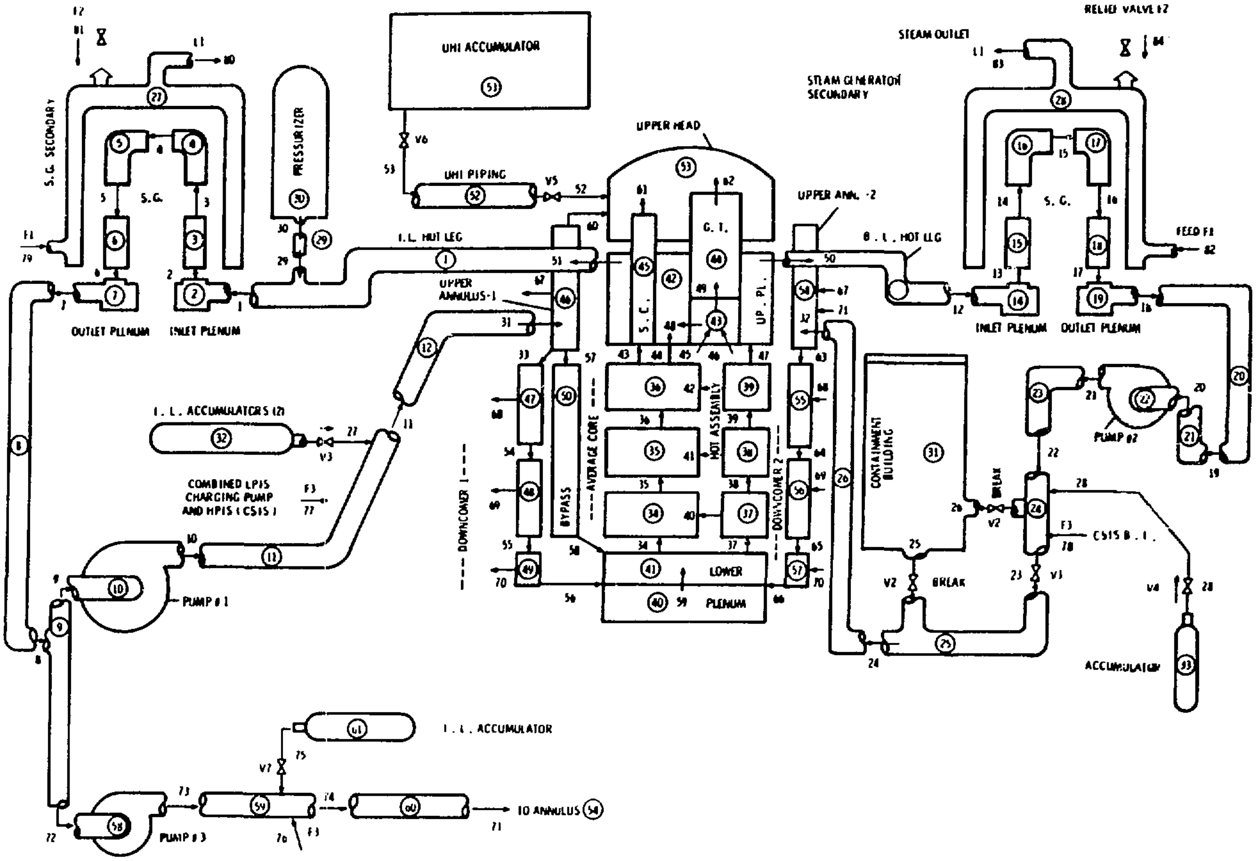


Figure 2. Nodalization for RELAP Blowdown Analysis

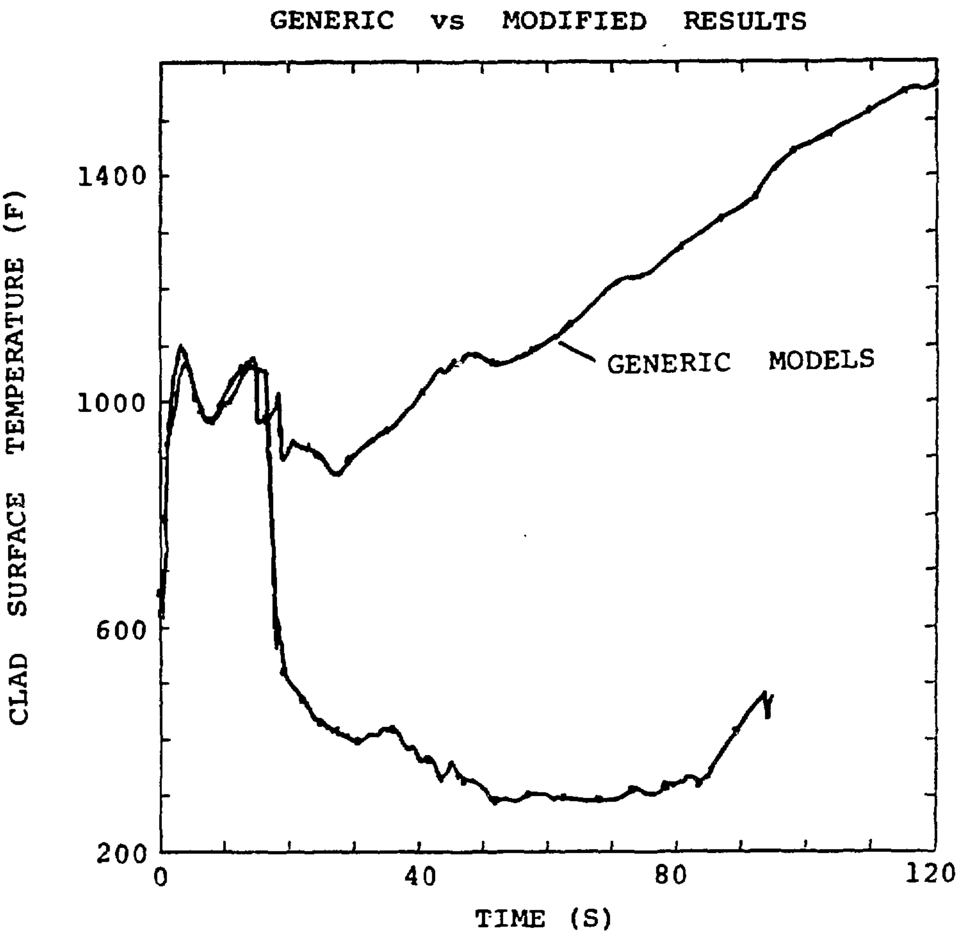
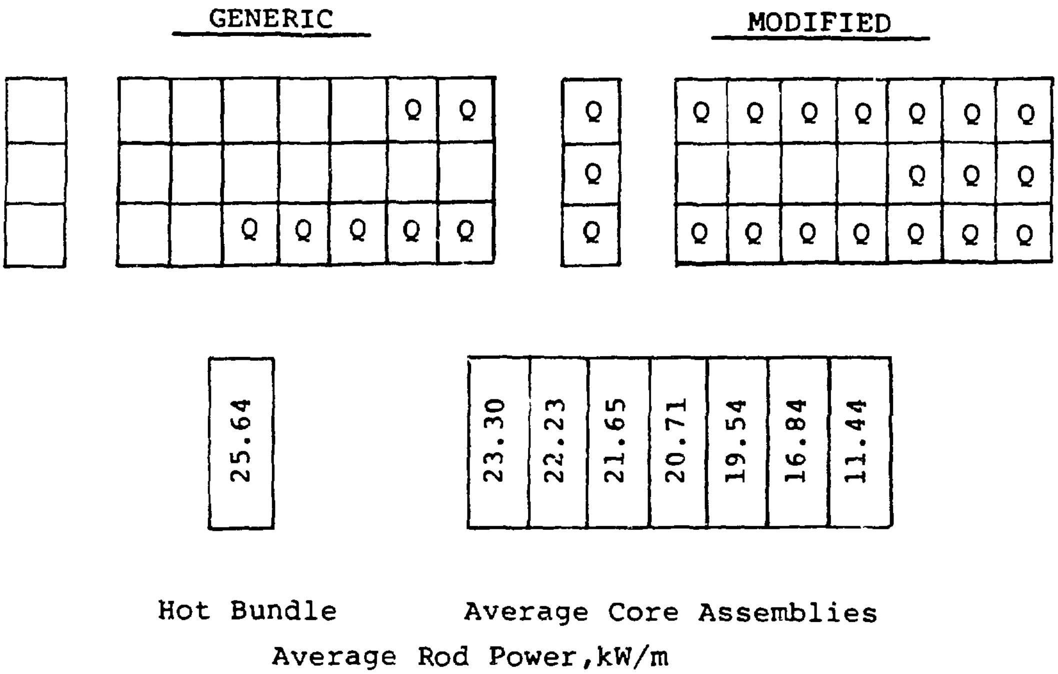

GENERIC vs MODIFIED RESULTS

Figure 3. Fuel Rod Surface Temperature at Midplane of Hot Bundle



GENERIC

						Q	Q
			Q	Q	Q	Q	Q

MODIFIED

Q	Q	Q	Q	Q	Q	Q	Q
Q					Q	Q	Q
Q	Q	Q	Q	Q	Q	Q	Q

25.64	23.30	22.23	21.65	20.71	19.54	16.84	11.44

Figure 4. Core Quench Maps at 80 s for Generic and Modified Results and Initial Average Rod Power

COMPARISON OF RELAP5 CALCULATIONS WITH LEVEL SWELL AND COUNTER-CURRENT FLOW PHENOMENA

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ABSTRACT

Independent calculations with RELAP5 have demonstrated an advanced state-of-the-art in thermal-hydraulic analysis. Level swell was investigated by analyzing a single vessel blowdown experiment. RELAP5 reasonably predicts void distribution without direct correlations for slip velocity or bubble rise models.

Countercurrent flow was studied by analyzing steady-state flow at atmospheric pressure in an 8x8 rod bundle. Initial RELAP5 results using the full flow regime map showed alternating periods of liquid downflow and vapor upflow.

By using only the annular flow drag correlation, a realistic flooding curve was generated by RELAP5. The flooding curve, as calculated by RELAP5, falls between the Wallis and Kutateladze correlations and was closer to the Wallis correlation at low gas fluxes, while being closer to the Kutateladze correlation at high gas fluxes.

Previous computer codes have used empirical flooding correlations, derived from unique experiments, whereas RELAP5 depends upon the phasic momentum equations to calculate the separate steam and liquid velocities.

INTRODUCTION

A fast running, two-velocity, nonequilibrium, thermal-hydraulic computer code, RELAP5/MOD"0" [1], has recently become available from the Idaho National Engineering Laboratory (INEL). RELAP5 shows the ability to analyze nonequilibrium, nonhomogeneous behavior of transient two-phase flow. This ability has been quite limited in earlier thermal-hydraulic codes such as those derived from the FLASH [2] series of codes. RELAP5 should, therefore, enhance the analyses of many LOCA phenomena such as level swell, counter-current flow limiting (CCFL) and refill and reflood hydraulics.

As an evaluation of the basic capabilities of the code, Intermountain Technologies performed independent RELAP5 calculations [3] of level swell and CCFL phenomena. These phenomena have been difficult or impossible to analyze with homogeneous, equilibrium codes of the FLASH or RELAP4 [4] type. The RELAP5 code is a two-velocity, two-temperature code and is not limited to the use of empirical correlations for the analysis of phenomena such as those discussed herein.

CCFL CALCULATIONS

The ability of RELAP5 to calculate CCFL phenomena was investigated by modeling an 8x8 fuel rod bundle. The RELAP5 model of the experiment is shown in Figure 1. The total length of the bundle is 2.43 meters. Of that length, 1.67 m extends below the tie plate and 0.76 m extends above the tie plate. Flow area is constant throughout the bundle except at the tie plate, which has a flow area of about 77% of the full flow area in the bundle. The large volumes at the top and bottom of the model represent sinks at nearly constant pressure. A constant liquid flow of 0.973 kg/sec was injected into the top of the bundle. Vapor was injected into the bottom of the bundle at various mass flow rates. The entire bundle was initiated at atmospheric pressure and 50% void fraction. Junctions were initialized at zero flow. Both liquid and vapor injection began at time zero.

The purpose of the CCFL calculations was to determine if the code would generate a smooth flooding curve. Separate phasic mass flow rates were monitored at the tie plate. Initial RELAP5 results showed alternating periods of liquid downward flow and vapor upward flow. In order to achieve steady, as opposed to alternating, counter-current flow, the flow was locked in specific flow regimes. Separate calculations were carried out with the flow regime fixed as annular, bubbly and dispersed.

Calculations with bubbly and dispersed flow regimes failed to show counter-current flow. In the bubbly regime, co-current downward flow was overcome by co-current upward flow at a vapor mass flux of 0.58 kg/sec-m2. The corresponding liquid flux was 50.7 kg/sec-m2. For dispersed flow the vapor and liquid mass fluxes at the point of turnaround were 0.53 kg/sec-m2 and 23.0 kg/sec-m2, respectively. For both bubbly and dispersed flow regimes, the flow direction reversed when velocities were near 1 m/sec.

Analysis of CCFL using the annular flow regime showed steady counter-current flow. Figure 1 shows the resulting flooding curve with comparisons to Wallis [5] and Kutateladze [6] correlations. The RELAP5 flooding curve lies between the two correlations. The RELAP5 calculation has better agreement at low gas fluxes with the Wallis correlation and at high gas fluxes with the Kutateladze correlation.

The ability of RELAP5 to produce a smooth flooding curve represents an advancement in the state-of-the-art. Previous thermal-hydraulic computer codes have used empirical flooding correlations, derived from unique experiments, whereas RELAP5 solves separate phasic momentum equations to calculate individual component velocities. Thus, more theory is intrinsic in the calculation, and reliance on empiricism is decreased. This more theoretical basis in RELAP5 holds the promise that calculations of full scale plants will show proper thermal-hydraulic trends.

Failure to obtain CCFL with the dispersed and bubbly flow regimes is quite realistic, since these regimes have high interfacial drag. The results showed very little difference in phase velocities in these flow regimes. The steady counter-current flow in the annular flow regime is due to less interphase drag in this high void regime, thus allowing the liquid to fall through the tie plate and the steam to release to the upper portion of the bundle. The alternating nature of calculations using the full flow regime map demonstrate the need to carefully compare RELAP5 results to a variety of flooding experiments. There is also a specific need to develop a correlation for interfacial friction in the slug flow regime. Future development should also focus on determination of proper transition points between drag correlations of the various flow regimes.

LEVEL SWELL CALCULATIONS

RELAP5 was next applied to a simple level swell experiment. The experiment was performed by the General Electric Company and was designated GE Level Swell Test 1004-3. Fischer and Hendrix [7] have compared RELAP4 calculations with the experiment. They reported that best agreement with the rate of depressurization was obtained with a discharge coefficient of 0.65.

Figure 2 shows the RELAP5 model of the level swell experiment. A 0.3 m diameter, 4.3 m high tank was initially filled with liquid to a level of 3.2 m. Initial pressure was 6.9 MPa. Discharge flow leaves near the top of the vessel through an orifice 0.0095 m in diameter.

Two calculations were performed. One used a discharge coefficient of 1.0. The other used 0.65 as recommended by Fischer and Hendrix for best comparison with the measured depressurization rates. Figure 2 shows the depressurization rates for the two transients. As expected the higher discharge coefficient results in more rapid depressurization. In both cases no liquid was carried out the discharge line. It was also apparent in both cases that liquid fallback was present at the top of the mixture.

The vapor void fractions at various elevations are compared with the experimental data in Figure 3. Both analyses showed higher void in the lower regions than did the data. In the mid and upper sections both analyses showed good agreement with the data. The results of the analysis with a discharge coefficient of 0.65 was farther from the data than the results with a discharge coefficient of 1.0.

The overprediction of vapor void fraction in the lower region of the vessel again demonstrates the need to develop an interfacial friction correlation for the slug flow regime. Future comparisons to a variety of level swell experiments could provide a method for determining transition points from one flow regime to another.

CONCLUSIONS

These calculations demonstrate the basic ability of RELAP5 to calculate two-velocity phenomena such as CCFL and level swell phenomena. The calculations of level swell exhibit CCFL phenomena at the mixture level. These results indicate that RELAP5 has applicability to analyses of LOCA hydraulics. While these results are preliminary, they show that future application of RELAP5 to thermal-hydraulics has strong potential. The RELAP5 code should continue to be compared with data from experiments covering a wide range of geometric and fluid conditions.

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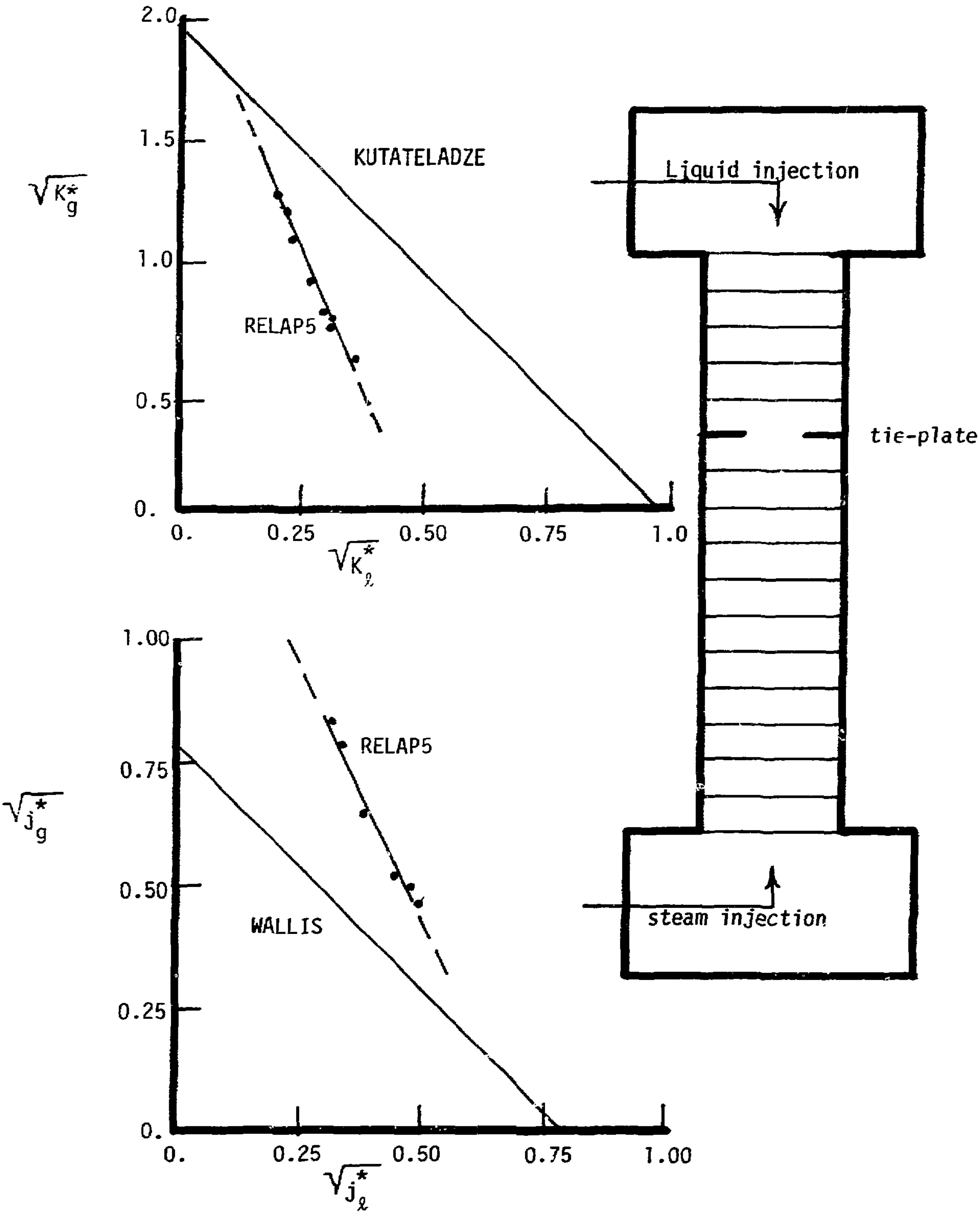

Figure 1. Comparison of RELAP5 to CCFL Correlations

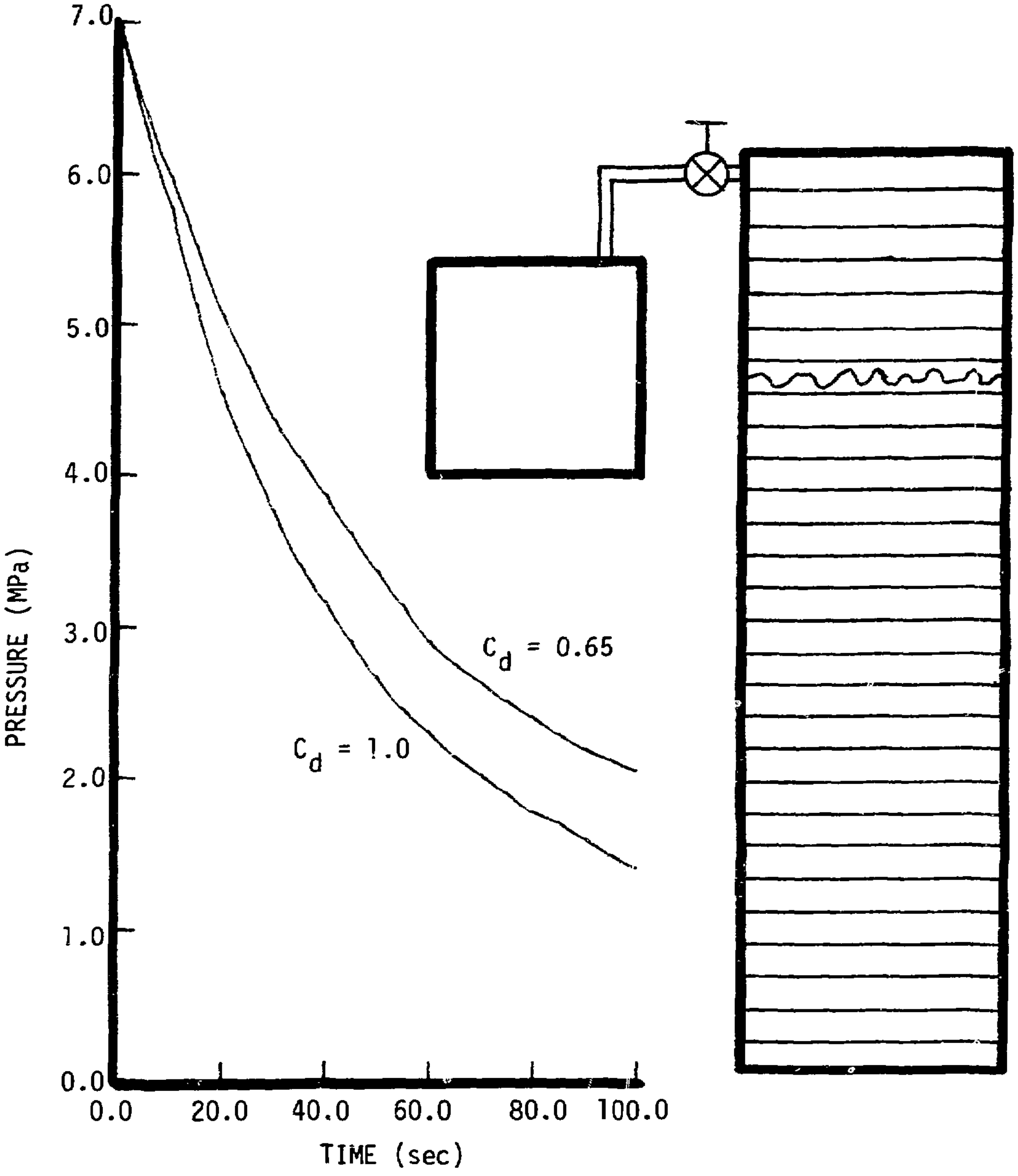

Figure 2. RELAP5 Depressurization of GE Level Swell Test 1004-3

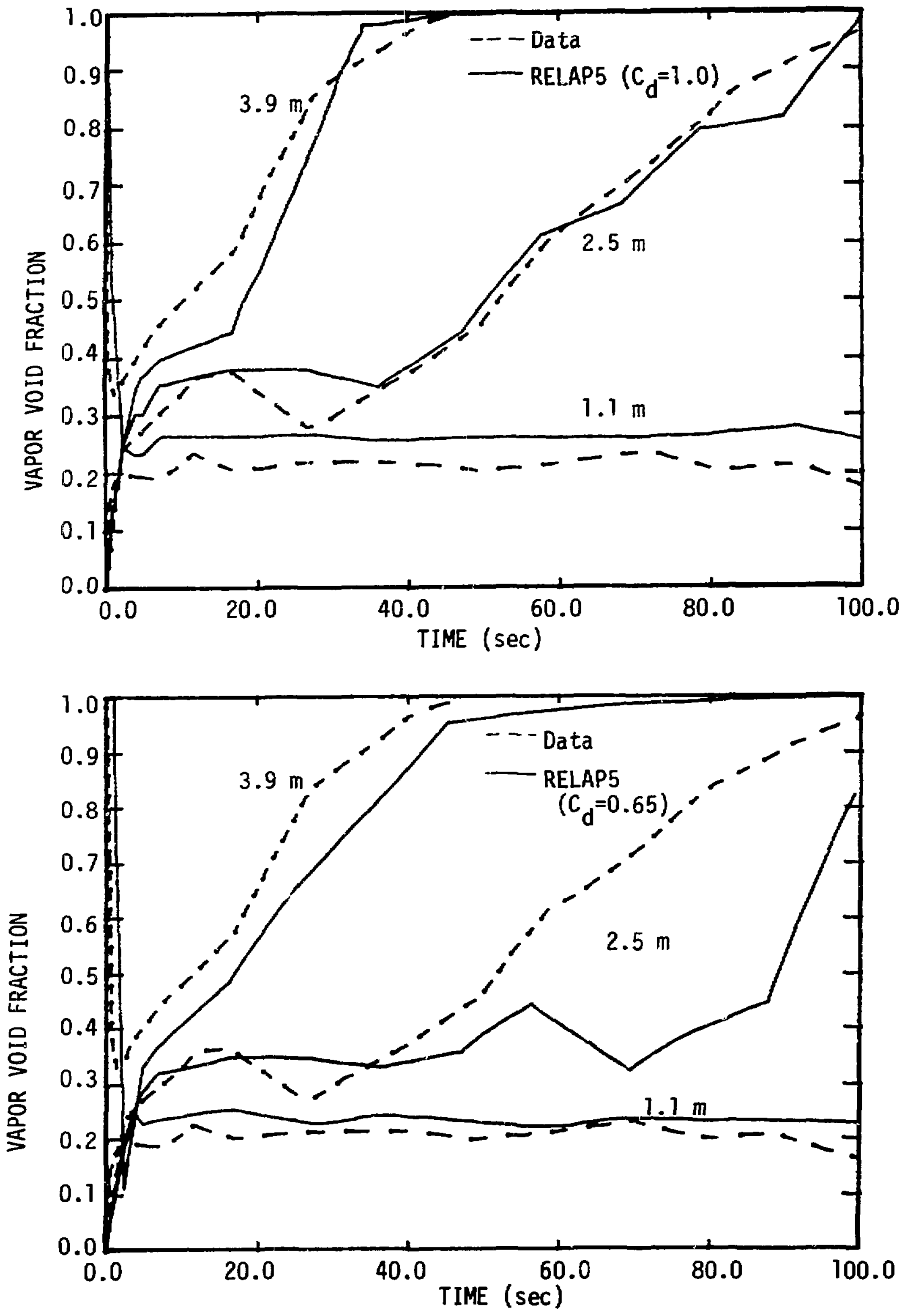

Figure 3. Vapor Void Fraction of GE Level Swell Test 1004-3

SEMISCALE SMALL BREAK ANALYSIS WITH RETRAN*

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ABSTRACT

The thermal-hydraulic codes in general use today for system calculations have been heretofore developed for analyses of loss-of-coolant accidents following the postulated rupture of a large coolant pipe. The RETRAN computer program,[1] which was developed from a RELAP code, can be included in this category.

These codes generally assume one-dimensional flow and homogeneous equilibrium (HEM) conditions. Phenomena such as phase separation and separate phase velocities are not accounted for in such a code except by the use of special models. Application of the Wilson bubble rise model to conditions of phase separation is one example. Such special models permit the use of these codes for the analyses of small break events. This paper reports application of the RETRAN program to analyses of a small break experiment (S-02-6, performed at the Semiscale facility). Two base calculations were performed, (1) an HEM calculation with the released version of RETRAN and (2) an analysis with a developmental version of the code which allows for separate phase velocity calculations.

RETRAN CODE DESCRIPTION

The released version of RETRAN[1] solves the homogeneous equilibrium hydraulic equations assuming one dimensional flow. From the viewpoint of a small break calculation, a major limitation is that the two phases are treated as though they move at the same speed. To overcome this deficiency, a version of RETRAN is being developed which uses a time-dependent equation to solve for the difference in velocity (slip velocity) between the two phases. A preliminary model for this equation has been discussed previously.[2] An extensive development effort was recently undertaken to extend the analysis capability of the dynamic slip equation.

The present slip model has been used to analyze steady-state void fraction and pressure drop experiments, the standard RETRAN sample problems, the Peach Bottom turbine trip tests; and currently RETRAN is being used to analyze the TMI-2 incident. The application of this code for analyzing a Semiscale small break test (S-06-6) is presented below.

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* This research was supported by the Electric Power Research Institute.

SEMISCALE TEST S-02-6 ANALYSIS

Semiscale Test S-02-6 was designated by the U.S. Nuclear Regulatory Commission as Standard Problem 6 in the Comparative Analysis of Standard Problems (CASP) program. This test was modeled prior to the release of the test data;[3] however, the actual initial test conditions were different than those assumed for the pretest calculation. The major difference was that the model assumed the steam generator isolation occurred at the time of the break rather than at 15 seconds into the transient, as in the test. Consequently, the former analyses predicted much higher system pressures and temperatures throughout the transient.

The model was revised to delay the steam generator isolation until 15 seconds into the transient, as in the test data. In addition, the model noding was revised in the broken loop, and several volumes in the core and steam generator primary side were combined to provide a simplified model for the initial slip model calculations. A schematic of the nodalization is shown in Figure 1.

A significant limitation to evaluating the code's present calculations is the amount of scatter and large error bands on the experimental data used in this study; see Reference 4. The data presented in this report were plotted from a magnetic data tape obtained from the experiment. Figure 2 shows a plot of the several break flow data measurements which were made during the test. As indicated on the figure, one curve was modified to change the sense of the flow to the positive direction. In a second case, a multiplier of four was used to provide a curve which is more consistent with the remaining data. The multiplier was used because the experimentor suggested[4] that a gain control on an instrument may have been improperly recorded. Note that during the initial 60 seconds of the transient, the five measurements differ by a factor of about three at any given time, ranging from a lower value of 2.5 lbm/s (1.1 kg/s) to upper values of 9 lbm/s (4 kg/s). Even at about 80 seconds and beyond, when the flow becomes high quality steam, the scatter among the various data measurements remains quite large.

The intact and broken loop high pressure injection system (HPIS) flows are shown in Figures 3 and 4, respectively. The error bands on the intact loop (Figure 3) HPIS flow, determined by the experimenter, amount to ±100% of the measured value. In addition, the measured signal contains a significant noise level prior to the initiation of flow, which begins at approximately 38 seconds. Although error bands were not supplied for the broken loop HPIS flow data, they are assumed to be of the same order because of the similar characteristics and the noise signal level. An additional modeling difficulty was encountered in the broken loop because the HPIS flow was injected immediately upstream from the break nozzle, and downstream from the measurement locations. Hence, the test configuration not only creates a difficult modeling and interpretation configuration; but because the injection location is very close to the break nozzle, significant nonequilibrium conditions may have existed in the critical flow region.

To provide a basis for evaluating the slip model, an HEM calculation was performed using the identical nodalization, boundary conditions, etc., that were used for the slip calculation. Note that the study performed and discussed in this paper is an initial attempt to evaluate the dynamic slip model, and that a very limited effort has been made to determine the optimum nodalization, and other model features required to ensure close agreement with the experimental data.

Three experimental data curves were selected for comparison with the predicted break flows. Two of the data were taken from measurement locations in the break line. A third curve was derived from the difference between the intact loop cold leg flow and the core inlet flow data, which would be equal to the break flow. This curve and one of the more direct measurements agree quite well during the first 60 seconds. The break flows for both HEM and SLIP calculations are in reasonable agreement with these data until 20 seconds into the transient when the predicted break flows begin to decay much more rapidly than the data. The predicted decay rate is reduced somewhat when the HPIS flows are initiated at 38 seconds into the transient. After 80 seconds, when the flow has become high quality two phase, the predicted values are somewhat larger than the measured data; but, the SLIP solution is showing better agreement.

The measured and predicted densities at two locations upstream from the break junction, shown in Figure 6, are in good agreement until 60 seconds when the fluid begins to drop below saturation density. The calculated density provides the most marked evidence of the capability of the dynamic slip model, as shown by the comparisons after 60 seconds into the transient. There is some delay in the time both calculated densities begin to decay. The HEM calculated density decays much more slowly than the SLIP model, and the value levels off considerably above the measured value. The slip calculated density initially decays at approximately the same rate as the experimental data and appears to be converging toward the data after 100 s. Nodalization optimization and/or dynamic slip model refinements may be required to obtain a closer agreement with the measured data.

The discrepancy in the characteristics of the predicted versus the experimental flow during the initial 60 seconds, indicates that either the measured values are not properly following the phenomena and/or the critical flow models are deficient in the characterization of critical flow. The critical flow model is based on HEM assumptions and may require modification to be consistent with the dynamic slip model.

The system pressure response is depicted by both the pressurizer and upper plenum pressures superimposed on Figure 7. Note that the pressurizer pressure decays slowly as the liquid mass is deleted from the pressurizer; whereas, the primary system pressure decays very rapidly to saturation. Once the pressurizer becomes voided, the difference in upper plenum and pressurizer pressures are indistinguishable on the scale shown. The HEM and slip calculations yield essentially identical pressure responses. The predicted values are in quite good agreement during the initial 20 seconds of the transient, when the break flow was also in reasonable agreement. During the 20-to-60 second time period, the predicted pressure is slightly higher than the measured value corresponding to the somewhat low predicted break mass flows. Hence, the net mass and energy in the model at 60 seconds into the transient is somewhat higher than that in the experimental system.

After 80 seconds into the transient, the predicted flows and densities were higher than the data. Hence, the predicted mass leaving the system is larger than measured; whereas, the energy leaving the system appears to be somewhat less than measured. Consequently, the predicted pressures are higher and appear to diverge from the measured values. From these results, one can see the importance of accurately predicting both the mass and energy release from the system during a small break.

The steam generator secondary response is shown in Figure 8. The temperature (saturation pressure) in the secondary remained essentially constant until approximately 14 seconds into the transient when the secondary was isolated by closing off the feedwater and steam line flows. Once the secondary system was isolated, heat energy continued to be transferred into the secondary causing the temperature (pressure) to rise until the relief valves opened at approximately 30 seconds into the transient. Later the primary system temperature has apparently dropped below the secondary temperature, the heat flow was then from the secondary into the primary system, and the secondary system temperature and pressure slowly decay thereafter.

The steam generator secondary was modeled using a single volume. The initial mass and energy in the secondary model appears to have been correct because the rate of temperature (pressure) rise immediately after isolation was in excellent agreement with experimental data. Thereafter, the predicted temperature continues to rise until the relief valve opened causing a depletion of mass energy in the secondary; once the low set point was reached the relief valve closed and the temperature began to rise again. The rising secondary temperature, after 50 seconds into the transient, indicates that the calculated primary temperature was also higher than the measured data, hence, the calculated heat flow was from primary to secondary, the opposite of the experiment.

Additional modeling studies are required to evaluate the effect of the secondary on the small break system response. In order to properly account for the effects of the natural circulation flow in the downcomer and bundle regions of the secondary, a multi-node model may be required. The current developmental version of the RETRAN code contains additional heat transfer correlations which are more appropriate, than those used in this study, for the low flow conditions which exist after the steam generator is isolated and the primary pumps have coasted down.

Three experimental data measurements of the lower plenum density were available for comparison to the calculations as shown in Figure 9. The slip calculation agrees quite well with the data; whereas the HEM calculation begins to diverge very rapidly at about 90 seconds into the transient. The core inlet density results, Figure 10, were similar to the lower plenum density because the core flow was positive upward during most of the transient.

The two predicted mass flows into the core are very similar. The meausred flow reversal during the 40 to 55 second time period was not predicted by the HEM or slip calculation (see Figure 11). The large difference in the HEM predicted fluid density entering the core could have a significant effect on the core response. In the case of these test data and predictions, the heater rod temperatures follow the saturation temperature within the system; thus they are of little if any interest in this evaluation.

The density at the intact loop pump inlet is shown in Figure 12. In this single case, the HEM calculation appeared to decay more rapidly than the slip calculation. During the 50 to 100 second time period the HEM calculation appears to be in better agreement with the data. Additional nodalization studies, improvement of the available heat transfer correlations and secondary modeling would be expected to improve the predicted response at this location and perhaps throughout the system as well.

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3. S. R. Behling, "Comparisons of the Standard Problem Calculations with Measured Experimental Data for Semiscale Test S-02-6", Idaho National Engineering Laboratory PG-R-76-004, December 1976.

4. B. L. Collins et al., Experiment Data Report for Semiscale Mod-1, Test S-02-6, (Blowdown Heat Transfer Test)," January 1977, TREE-NUREG-1037.

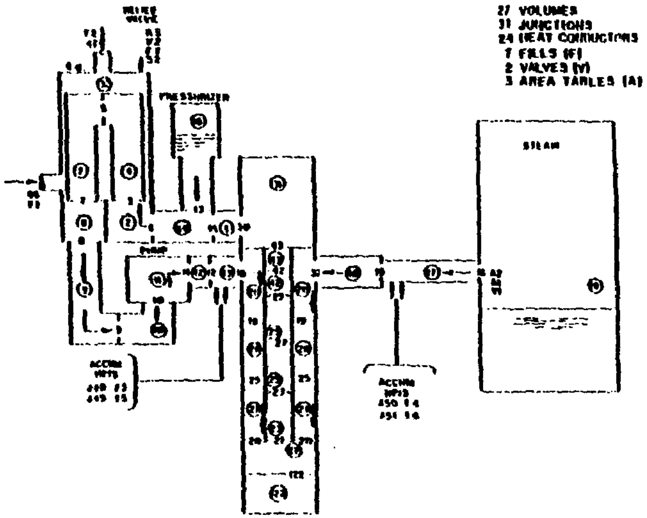

Figure 1. RELAP4 Model for Semiscale Test S-02-6 (Small Break)

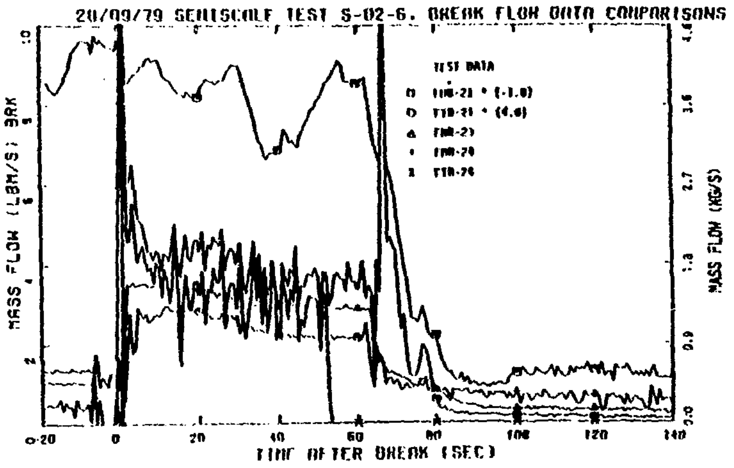

Figure 2. Break Mass Flow Measurements

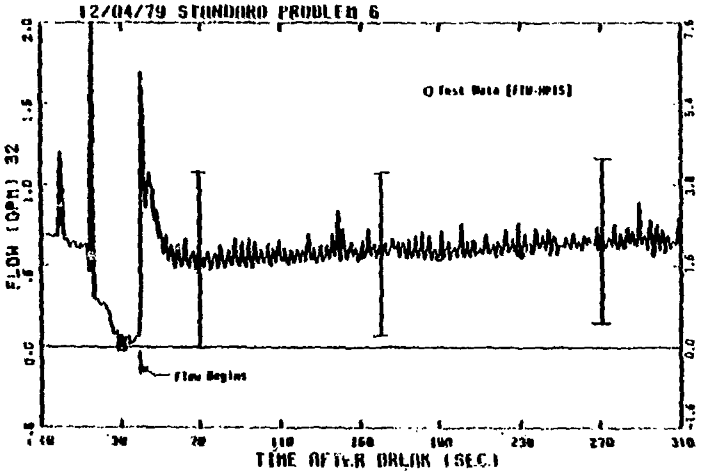

Figure 3. HPIS Flow - Intact Loop

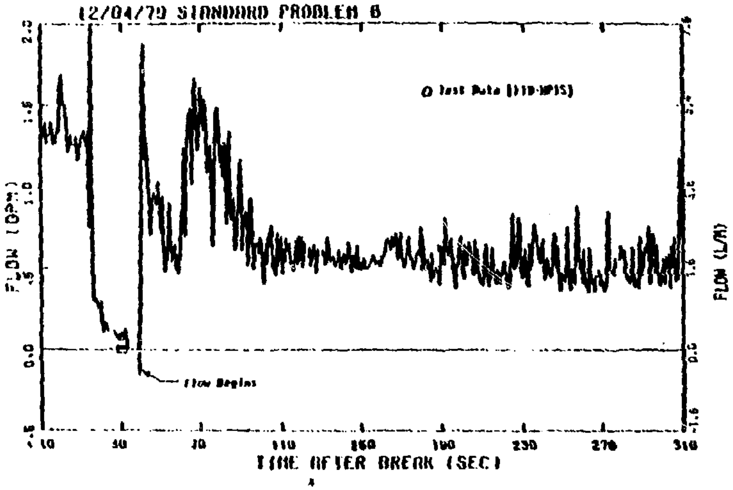

Figure 4. HPIS Flow - Broken Loop

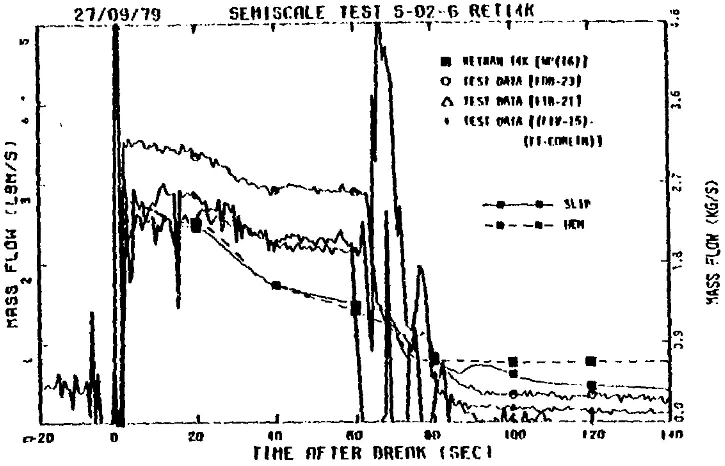

Figure 5. Break Mass Flow

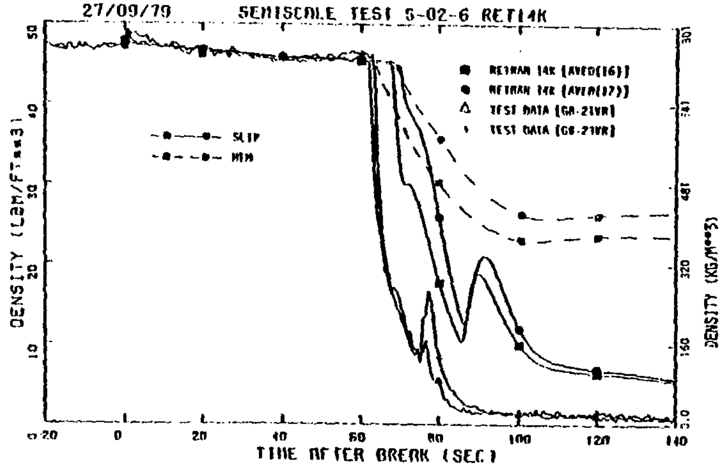

Figure 6. Break Flow Density

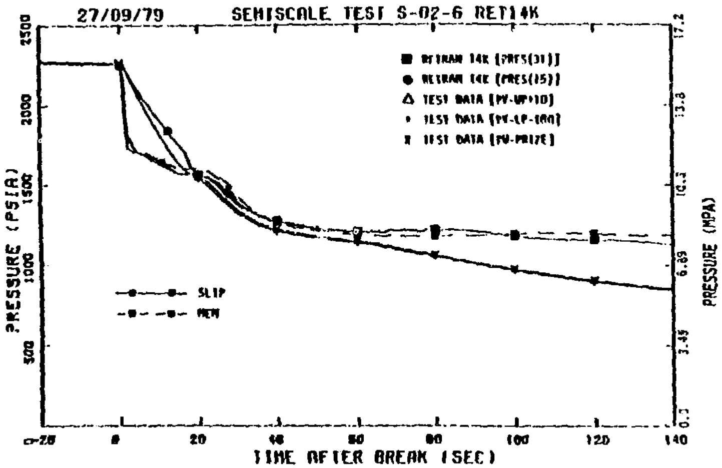

Figure 7. Primary System Pressure

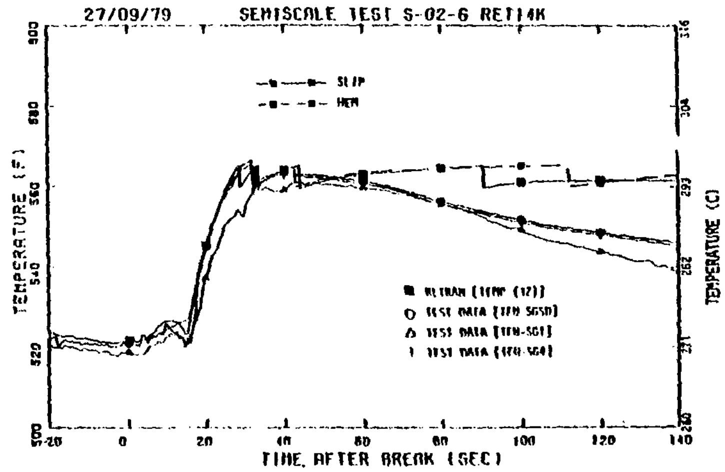

Figure 8. Secondary Temperature

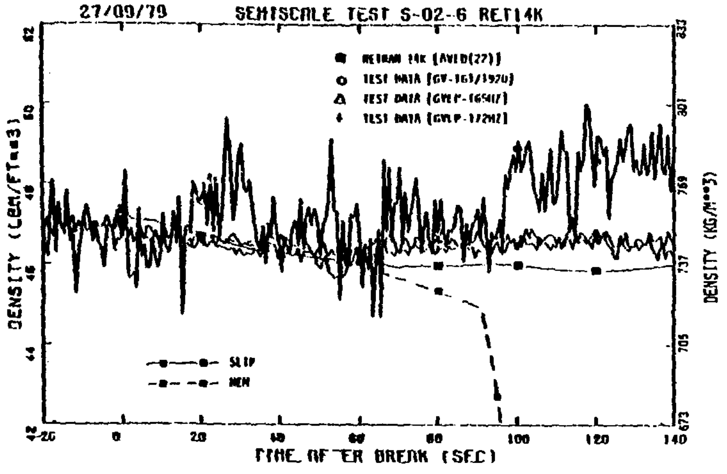

Figure 9. Lower Plenum Density

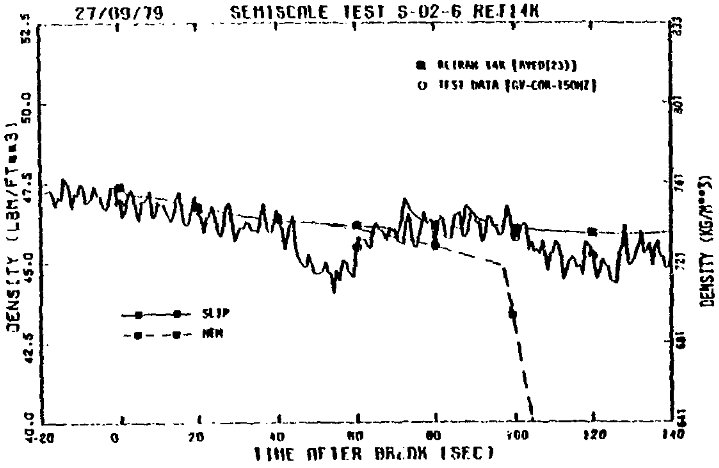

Figure 10. Core Inlet Density

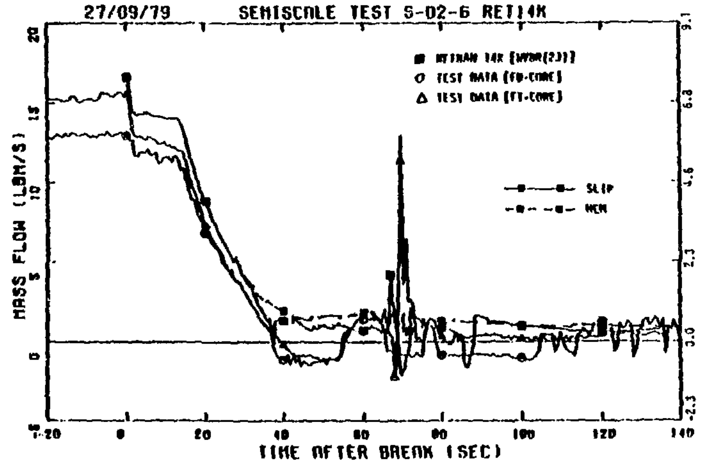

Figure 11. Core Inlet Flow

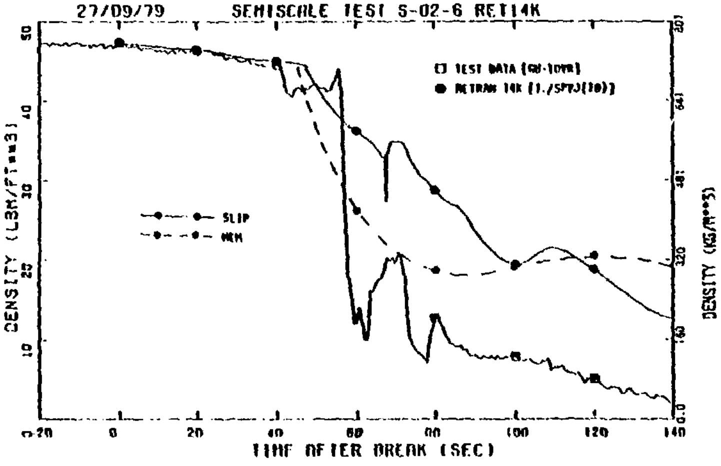

Figure 12. Pump Inlet Density

POST ACCIDENTAL SMALL BREAKS ANALYSIS

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ABSTRACT

EDF ordered to Framatome by 1977 to complete post accidental long term studies on "First Contrat-Programme" reactors, in order to demonstrate the safety criteria long term compliance, to get information on NSSS behaviour and to improve the post accidental procedures. Convenient analytical models were needed and EDF and Framatome respectively developped the AXEL and FRARELAP codes. The main result of these studies is that for the smallest breaks, it is possible to manually undertake cooling and pressure reducing actions by dumping the steam generators secondary side in order to meet the RHR operating specifications and perform long term cooling through this system. A specific small breaks procedure was written on this basis. The EDF and Framatome codes are continuously improved ; the results of a French set of separate effects experiments will be incorporated as well as integral system verification.

INTRODUCTION

Starting in 1975, a French programm for building 28 PWR nuclear power plants (900 MW, 3 loop plants) has been carried out by EDF.

Within the frame of this program, in addition to satisfy French Safety Authorities requirements, studies have been requested by EDF to Framatome, in order to estimate a long term behaviour of the plant in addition of the studies already done in the Safety Analysis Reports demonstrating the Appendix K requirements compliance.

The post accidental studies which started in 1977 have been carried on, and since 1979, they have taken into consideration the Three Mile Island Unit 2 accident lessons. They investigate any accident for which the cause remains unresolved after the automatic safety systems actuation. Small and intermediate breaks are involved in such accidents.

The objectives of these studies can be summarized as follows :

- to verify that the safety criteria are always satisfied until a safe and quasi steady state has been reached with minimum operator intervention ;
- to give informations about the main plant parameters in order to improve the diagnosis abilities and the understanding of the plant behaviour during the accident, more specifically the different cooling modes and the changing from one mode to another ;

- to allow improvements and justifications for the operational procedures, especially manual actions on steam generators, actuating hot leg injection, actuating RHR.

The studies presented in this paper mainly result from Framatome computations in 1978, and from a few cross-checking computations on some specific problems performed by EDF.

EDF DEVELOPMENTS

EDF uses two types of computer codes :

- SIBEL : a simplified code where primary and secondary sides are both simulated by one or two control volumes,
- AXEL : a code where primary and secondary sides are both simulated using one dimensional model (3 main equations).

The computations performed with these codes agree with the conclusions of Framatome studies ; they especially permit to specify the possibilities of cooling and depressurizing the primary by dumping the secondary side of the steam generators in case of small breaks.

FRAMATOME DEVELOPMENT OF A COMPREHENSIVE ANALYTICAL MODEL

The different aspects of the long term cooling to be evaluated are the pressure transient, the recirculation procedures and time, the behaviour of the steam generators during the long term, the inventories of steam and water within the reactor pressure vessel, the possibilities of boron concentration built up in the core, the containment heat removal.

Since the small break spectrum covers equivalent break sizes of less than 2,5 cm (1 inch) up to about 25 cm (10 inches), break area ratio of more than one hundred, it is expected that the long term transients and behaviours may vary within large scales.

First of all, it was necessary for Framatome to develop an acceptable tool for small break thermal hydraulic transient analyses (FRARELAP).

Early in 1977, it was decided to evaluate the possibilities of the RELAP 4 MOD3 code. It appeared that this computer code could be the acceptable tool if some modifications are done. We performed these modifications during the two and a half past years.

The purpose of these modifications is threefold :

- better analytical models for small break transients,
- user's conveniences,
- computer code running time reduction.

Provided are the main aspects of the RELAP 4 MOD3 improvements for small break analyses :

- bubble rise velocities function of both pressure and void fraction based on separate effect experiments,
- heat transfer coefficients between the coolant and heat slabs, core or metal heat, depending on the mixture level,
- possibility of the core node to have a mixture level and a superheated steam dome ; steam superheats along the uncovered core,
- gravity heads based on steam water interface as reference,
- heat transfer coefficients in the steam generators based on covered and uncovered parts, as for heat slabs, with in addition test of the heat flux direction : steam generators secondary cooling or heating the primary side,
- non horizontal flow paths, T nodes, safety or relief valve, pump modeled in a flow path, trips to stop a function, additionnal outputs...

POST ACCIDENTAL STUDIES RESULT

The main results of the long term post accidental studies performed with FRARELAP are described in the following.

The small breaks spectrum can be split into three different types :

a) intermediate breaks of sizes larger than 10 cm (4 inches),
b) intermediate breaks of sizes less or equal to 10 cm (4 inches),
c) small breaks of sizes less or in the range of 2,5 cm (1 inch).

a) For the first ones, the break size is rather large, the system depressurizes rather quickly even without any operator action to dump the secondary side of the steam generators (figure 1).
The steam generators primary side tubes experience quick draining ; natural circulation fastly interrupts and never gets reestablished.
Residual heat is removed through the break only when breakflow turns from saturated liquid to saturated steam, after clearing the loop seal in case of a cold leg break.
Steam generators are no longer needed for the remaining of the transient. At the cold leg recirculation time, the low head safety injection pumps may or not inject directly into the reactor coolant system. Thus, the recirculation through the high head safety injection system is needed.
Steam is continuously escaping out of the core and boron concentration built up may occur. Switchover from cold leg recirculation to hot leg recirculation is needed at about 18 hours.
At such long term, the low head safety injection provides enough flow into the reactor vessel.
Since for such break sizes, the steam generators are of no aid in cooling the system, the whole core residual heat is released to the containment through the break.
The containment pressure and temperature are reduced by the containment spray system ; the heat exchangers on the spray system are used to cool the containment sumps.

b) For the smallest intermediate break sizes, about 5 cm (two inches) equivalent diameter, similar phenomena occur.
For about 250 seconds, natural circulation keeps removing the residual heat through the steam generators with a primary flowrate of 10 % of nominal flowrate, system pressure being controlled by pressurizer and next upper plenum.
As break size is still large enough, safety injection flow does not match the liquid break flow and the system empties.
The previous natural circulation modes stop when the mixture level in the pressure vessel falls below the upper elevation of the hot legs. From this time, the upflow part of the steam generators U-tubes start draining, and next the down flow part. The driving head reduction decreases the loop flow.
The mixture level in the core keeps going down and in case of a cold leg break, core uncovery may occur until the downcomer driving head matches, the pump suction resistance head. No credit for countercurrent flow in hot leg is taken.
At about 1200 seconds, the loop seal clears and breakflow turns from saturated liquid to saturated steam ; residual heat is mainly removed through the break.
As safety injection flow matches the saturated steam break flow, no further core uncovery may occur but steam production in the core is still going on.

Without dumping the steam generators secondary side, the primary pressure remains above the low head safety injection pumps shut off head for about 48 hours (figure 2). Cold leg recirculation using the high head pumps is required. Again the sumps have to be cooled through the heat exchangers on the containment spray system. The primary system is almost in a steady state condition with full downcomer and core covered (figure 3) ; the safety injection flow matches the boil-off flow ; residual heat is removed through the break to the containment. Hot leg recirculation on the long term using the high head pumps prevents from core boiling.

With dumping the secondary side, the condensing heat transfer coefficients allow a faster depressurization of the primary system (figures 4 and 5).

At the switchover time from cold leg recirculation to hot leg recirculation, i.e. about 18 hours, pressure may have decreased enough so that low head safety injection pumps inject into the primary. Safety injection flow then matches the liquid break flow, the primary system refills and may go from core boiling to natural circulation.

c) The small breaks, about 2,5 cm (1 inch) equivalent diameter, behave differently. Early in the transient the safety injection flow is able at high pressure, secondary valve setpoint or more, to match the break flow. Residual heat cannot be removed through the break because of its small size and the steam generators cooling is needed for an extended period of time.

The steam generators auxiliary feedwater system keeps the secondary water level at the 0 % power level and compensates for the steam flow released to the condenser or to the atmosphere.

The primary system remains essentially in liquid, and does not depressurize if no operator action is taken (figure 6) ; natural circulation controlled by upper plenum pressure keeps removing the residual heat to the steam generators for about 24 hours if no operator action (figures 7 and 8).

For these break sizes, primary pressure above secondary pressure, the operator may dump the secondary side at a rate of 28°C/hour. Heat transfer from the primary to the secondary allows to depressurize and cool the primary system. After about 3 hours cooling, the primary parameters are as follows :

- pressure about 3 MPa (figure 9),
- temperature less than 180°C (figure 10),
- injection flow matches the subcooled break flow,
- the system is solid, with visible level in the pressurizer (figure 11).

Therefore, since these conditions are equivalent or very close to the RHR system operating specifications, the long term cooling may continue using this system. As a benefit, this procedure may allow not to actuate the containment spray system.

POST ACCIDENTAL PROCEDURES

The previous demonstration made it possible to write a suitable post accidental procedure for small breaks.

This procedure tells the operator to use the normal systems - i.e. the RHR - for long term cooling after appropriate actions.

The symptoms allowing the operator to undertake cooling and pressure reducing actions by dumping the steam generators secondary side, in order to meet the convenient conditions of pressure and temperature in the primary for turning on the RHR, are the following ones :

- slow decrease of pressurizer level and pressure,
- primary pressure equilibrium above steam generators secondary pressure, if no fast cooling action,
- efficient cooling of primary through steam generators secondary side (condenser or atmosphere),
- slow increase of containment pressure,
- identical pressure in each steam generator secondary side,
- core outlet thermocouples showing saturation or subcooled conditions.

However, if any reason such as equipment unavailability makes this procedure fail, the operator uses the large or intermediate breaks post accidental procedure, that is cold leg recirculation, switchover to hot leg recirculation at about 18 hours after the accident and sump water cooling through containment spray exchangers.

FUTURE DEVELOPMENTS AND ORIENTATIONS

Independently of the previous long term studies performed during 1978 and 1979, new studies were ordered following TMI accident, dealing with pressurizer vapor space breaks, effects of primary pumps running, break isolation, non condensible gases effects, safety injection termination, breaks occuring during cold shutdown with RHR in service.

For these purposes, the FRARELAP and EDF/AXEL computer codes are continuously improved. Results of separate effects experiments have been or will be incorporated as well as integral tests verifications based on LOFT or LOBI.

Separate effects experiments are conducted in cooperation between Westinghouse, CEA, Framatome and EDF. They are briefly listed in the following table (see under).

Specifically, PWS 2.21 (Westinghouse proprietary tests on the G2 loop facility) results were used as a basis of improvement of bubble rise model in FRARELAP code.

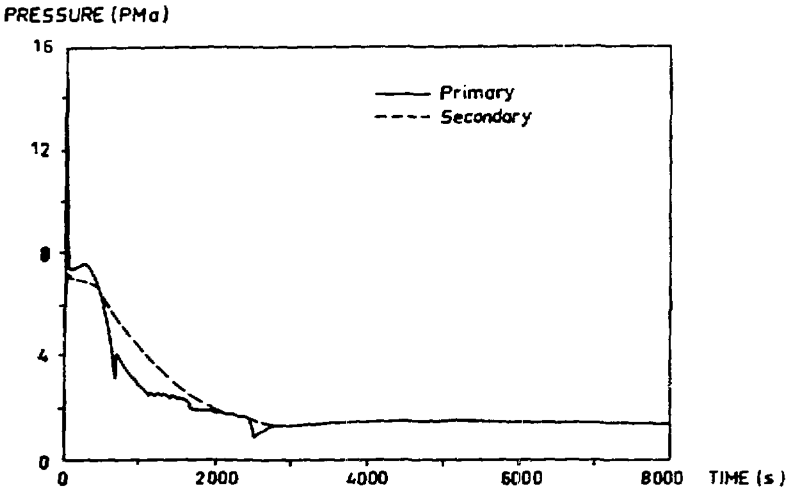
PWS 2.3 is an experiment of horizontal pipe flow regime in large diameter and simulates both the U-tube loop seal and the hot leg (results expected in mid 1980).

PATRICIA is a separate effects experiment for studying the heat transfer coefficients and hydrodynamic behaviour of the steam generator in degraded situation (first results expected in 1981).

On this specific point, EDF/AXEL code allowed to define the boundary conditions required to develop this experiment (temperature, void fraction, flow...).

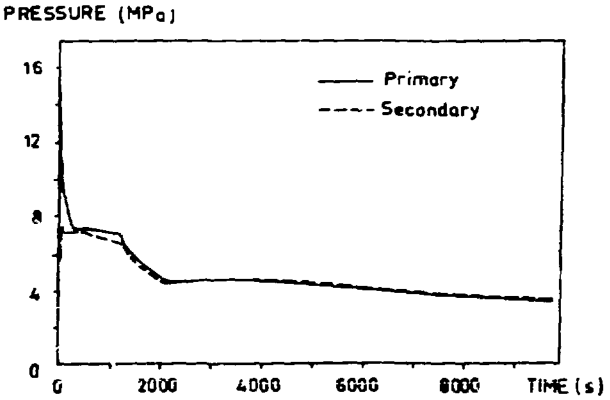
At last, a joint team composed by CEA, EDF and Framatome is developing a new generation code using a 6 equations model for both small and large breaks analyses.

SMALL BREAKS		MASS & MOMENTUM TRANSFER				WALL HEAT TRANSFER		COMPONENTS					
SEPARATE EFFECTS EXPERIMENTS IN FRANCE	FIRST RESULTS	LOW VERTICAL VELOCITY	LOW HORIZONTAL VELOCITY	ENTRAINMENT - DEENTRAINMENT	NON CONDENSIBLE GAS EFFECTS	CORE	CONDENSATION IN S-G TUBES	UPPER PLENUM	HOT LEG	STEAM GENERATOR INLET & OUTLET PLENUM	STEAM GENERATOR 'U' TUBES	CROSS OVER LEG	PIPE CONNECTING JUNCTIONS
ECTHOR (PWS 2.3)	79,80	☒	☒					☒	☒	☒		☒	
PATRICIA 1	81	☒			☒		☒				☒		
SUPER MOBY DICK	81	☒	☒										☒
DADINE	81	☒											
REBECCA	81	☒			☒								
SEROPS	80			☒				☒					
SERPAT	79					☒							
ERSEC	80					☒							
SIROCCO	80					☒							
PERICLES	84					☒							



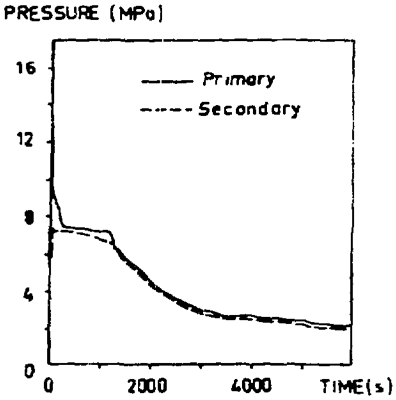
10 cm (4 inches) cold leg break

Figure 1



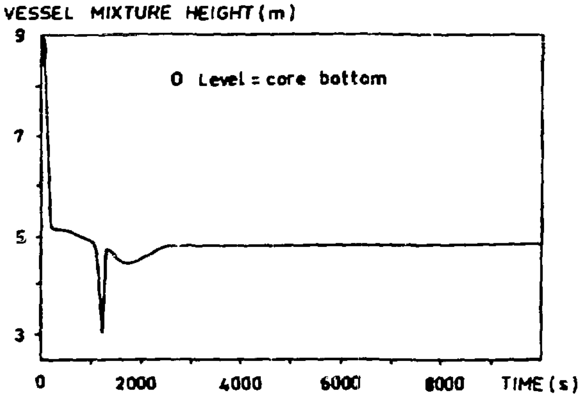
5cm(2inches) cold leg break
no operator action

Figure 2



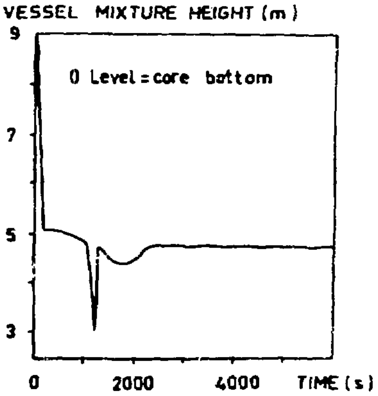
5cm(2inches)cold leg break
operator action

Figure 4



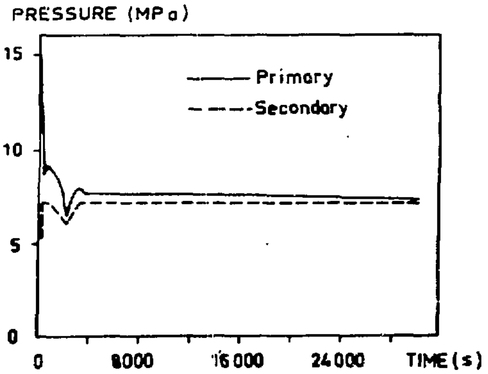
5cm (2 inches) cold leg break
no operator action

Figure 3



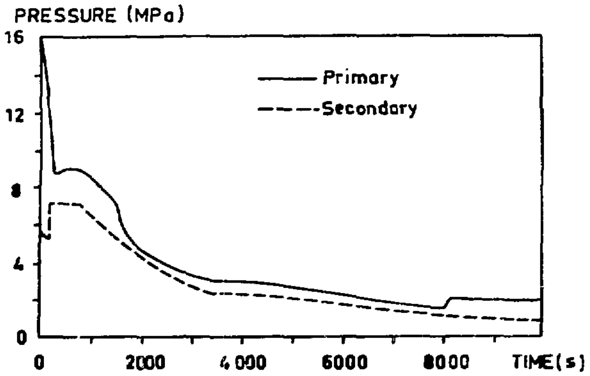
5cm (2 inches) cold leg break
operator action

Figure 5



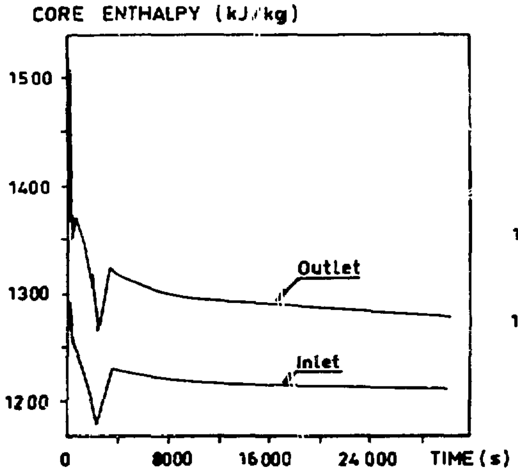
2,5cm(1 inch) cold leg break
no operator action

Figure : 6



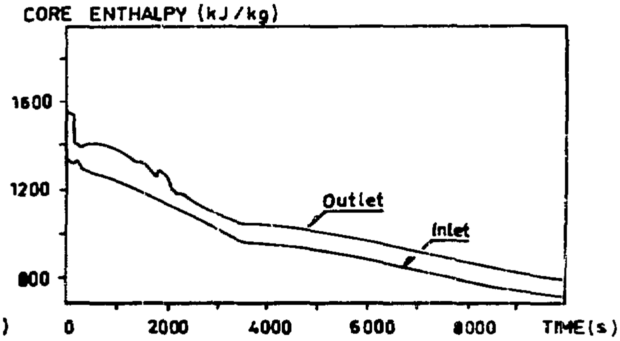
2,5cm(1 inch) cold leg break
operator action

Figure : 9



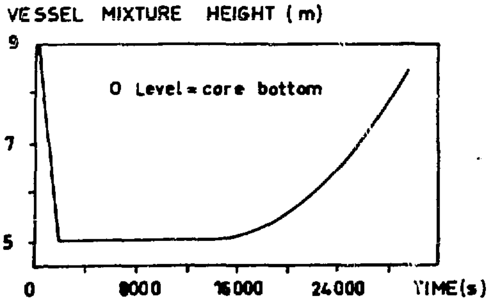
2,5cm(1inch) cold leg break
no operator action

Figure : 7



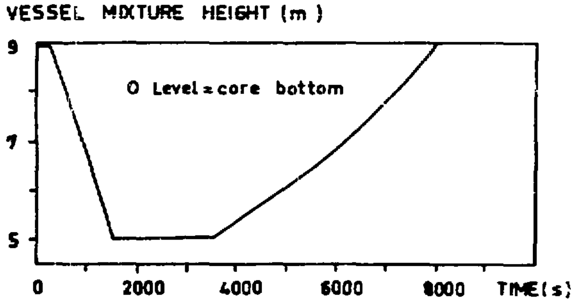
2,5cm(1inch)cold leg break
operator action

Figure : 10



2,5cm (1inch) cold leg break
no operator action

Figure : 8



2,5cm(1inch) cold leg break
operator action

Figure : 11

STATISTICAL ANALYSIS OF THE BLOWDOWN PHASE OF A LOSS-OF-COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR AS CALCULATED BY RELAP4/MOD6*

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ABSTRACT

A statistical study of the blowdown phase of a design basis loss-of-coolant accident (double-ended cold leg guillotine break) has been performed for the Zion pressurized water reactor using the RELAP4/MOD6 thermal hydraulics computer code. Twenty-one variables were selected for the study, including eight related to fuel behavior, five to heat transfer from clad to fluid and eight to two-phase flow and system parameters. The fuel variables dominated the predictions of peak clad temperature (PCT) as determined by response surface methodology. Based on the change in PCT per standardized change in the input variable, the most important variables were gap width, peaking factor and fuel thermal conductivity. Four more parameters were also found to be important. In approximate order, they included Condie-Bengston film boiling heat transfer, two-phase friction, phase relative velocity (slip) and power level. Critical flow and departure from nucleate boiling were not as important as these seven.

The RELAP data base was also employed to study other dependent variables besides PCT. Maximum oxidation depth, fuel stored energy, the rate of removal of stored energy and various core flow variables were investigated. These variables frequently behaved quite differently from PCT and may provide additional information for accident analysis.

INTRODUCTION

In NRC licensing procedures, plant safety is promoted by requiring that analytic models be "conservative" in the sense that they predict the worst of a set of possible consequences. These individually conservative models are collected in large computer codes to produce "evaluation models" intended to pessimistically predict the consequences of a variety of plant accidents. This approach has two possible weaknesses: First, although it is usually possible to demonstrate the conservatism of individual models, the complex physical interactions between various models may produce results which are not necessarily

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*This work was supported by the United States Nuclear Regulatory Commission.

"worst cases"; and second, it is frequently impossible to quantify the degree of conservatism in the evaluation model.

Studies have been supported at Sandia and other laboratories to investigate statistical methods for the analysis of reactor safety.[1-10] These methods have some important advantages. Probabilistic statements can be made concerning the results, thus permitting numerical estimates of the degree of conservatism. Another advantage is the utilization of "best estimate" rather than "evaluation model" codes. The accuracy of such codes can be assessed by comparison of their predictions with experimental data. A serious disadvantage is the necessity of performing a relatively large number of expensive calculations. We have recently completed a statistical study of the blowdown phase of a design basis accident (double-ended cold leg guillotine break) in the Zion pressurized water reactor.[1] The response surface method was employed to generate polynomial approximations of peak clad temperature and other core damage estimators as calculated by RELAP4/MOD6.[11, 12] The nodalization was a modification of the RELAP model of Zion developed in the BE/EM study.[13]

Twenty one variables were initially selected for the study. These variables, their ranges and distributions resulted from the best engineering judgment of NRC, Sandia, INEL and other interested and knowledgeable investigators.[1-10] Eight variables were related to fuel behavior and included reactor time-in-life, power, peaking factor, fuel thermal conductivity, gap width, decay heat, fuel swelling and blockage and metal-water reaction. Because of small sample size and other analytic problems, metal-water reaction rates were not included in the PCT response surface or distribution, but were analyzed separately. Time-in-life was employed in calculating the PCT probability distribution through its effect on gap width and peaking factor. It was not considered an independent variable in the response surface approximation.

Five variables were selected to characterize the heat transfer from the clad to the fluid. These were: critical heat flux; Condie-Bengston high flow film boiling; free convection and radiation; Dittus-Boelter reverse heat transfer from the fluid to the clad; and HSU and Bromley-Pomeranz low flow, low void fraction heat transfer.

The remaining eight variables included single- and two-phase flow parameters and miscellaneous ECCS-related quantities. These were: sub-cooled (Henry-Fauske) and saturated (HEM) discharge coefficients (two independent variables); churn-turbulent slip correlation (as implemented in RELAP4/MOD6); two-phase friction and form loss factors; containment pressure; ECC system temperature; two-phase pump degradation and accumulator pressure.

Approximately 200 RELAP blowdown calculations were performed during the study, including 153 using the MacDonald-Broughton gap conductance model, 26 with the Ross-Stoute gap conductance model and many preliminary sensitivity calculations.

RESPONSE SURFACE ANALYSIS OF PEAK CLAD TEMPERATURE

Twelve different PCT response surfaces were produced based on different underlying statistical assumptions. Since these assumptions are generally arbitrary, it is encouraging that the different surfaces yielded similar results. Table I summarizes the relative importances of the seven variables for a particular one of the response surfaces. The information is presented in two ways: the number of degrees change in PCT for a standardized change in the underlying input variable (approximately ± 1 usually); and the change in PCT due to a 1% change in the underlying variable above and below nominal. Based on 1σ changes, gap width was the most important variable. In terms of percent change, however, the dominant variable was power level. Note also that the sensitivity of PCT to input variable changes can differ considerably, depending on whether the change is above or below nominal.

Based on 1σ changes, the study indicated that these seven variables dominated the prediction of peak clad temperature. The three most important parameters were gap width, total peaking factor, PF, and fuel thermal conductivity, K. The four additional variables which were also found to have appreciable influence on PCT were, in order, film boiling heat transfer (CB-HT), two-phase friction, slip coefficient and power level. Critical heat flux and subcooled discharge coefficient were not as important as these seven. Evidence was produced, however, which implied that subcooled critical flow was more important for low values of PCT than for high. This effect can be seen in Figure 1. Notice only a slight difference in PCT observed with multipliers of 0.9 (nominal) and 1.2 on the Henry-Fauske critical flow model. A much larger difference occurs for the lower multiplier, and a quench takes place later in the transient. Since the statistical sample was intentionally biased toward higher temperatures, the reduced significance of subcooled discharge might, in part, be due to the smaller number of calculations at low temperatures.

The fact that peaking factor was more important than power level is due to the much larger range assigned to it. PF varied from 24% to 132% above core average power, while a ±3σ range for power level was ±6%. Since PF varied approximately ±30% about its midrange, it could be expected to be about 5 times as important as power level. This assumption is supported by the data in the table.

Because of a generic code error discovered late in the study, only a small number of calculations were performed in which the metal water reaction rates were varied. Although not included in the FCT study, these calculations were analyzed separately. Table II shows the changes in PCT observed for changes in MWR rates at various temperatures. The table indicates that increasing the MWR rates only begins to affect PCT when it is already high; e.g., using nominal rates (Cathcart-Pawel) yielded a temperature of 2185 for a particular calculation. Increasing the rate by 15% (+3σ) increased the temperature by 82°F. At lower temperatures, however, changes in MWR had little if any effect on PCT.

The seven variable PCT response surfaces were sampled 10,000 times to determine the estimated probability density functions and accumulated distributions. Figure 2 illustrates the PCT probability distribution for one of the response surface models. The sensitivities of the PCT distribution to changes in input means and variances were also computed. Table III shows the input mean sensitivities for the same model used in the earlier comparisons. This table, as expected, is very similar to the non-stochastic relative importance table.

RESPONSE SURFACE ANALYSIS OF OTHER CORE DAMAGE ESTIMATORS

PCT has traditionally been the most important measure of core damage for licensing applications. It should not, however, be considered the only such indicator. Other damage estimators can be defined which would enhance our knowledge of accident behavior, supplementing PCT and perhaps even replacing it under certain conditions. Damage estimators can be divided into two categories, local and global. Local estimates generally provide information on the state of that part of the core which has suffered the maximum damage during the course of an accident. PCT is in this class. The maximum oxidation of the rod cladding OD(t), would be another example. OD has the additional feature of being less susceptible to temporal fluctuations, such as quenching. OD is also an integrated quantity, monotonically increasing with time. As such, it may prove easier to model than PCT, which usually undergoes two distinct maxima, one during blowdown and the other during reflood. The total energy remaining in the fuel as a function of time FE(t), is an example of a relative global damage indicator. It provides information on core response which supplements PCT and OD, but would still be useful even for partial meltdowns, where PCT and OD become meaningless.

Response surfaces have been generated for PCT, OD at 20 s, and FE at 10 and 20 s. PCT and OD behavior were found to be quite similar. FE, however, automatically eliminates peaking factor as an important variable. Subcooled discharge coefficient also becomes significantly more important. The response surfaces for FE were particularly well behaved and predicted the RELAP data set with higher accuracy than was seen for PCT and OD predictions. All three dependent variables demonstrated that 129 data points were sufficient to predict the eight most important independent variables ($\sim$ 16 points per variable).

Analysis of the surfaces for PCT, OD and FE indicated that fuel variables were highly influential in determining ultimate core damage. This infuence was primarily exerted through the initial conditions prevailing at the time of the LOCA. To better understand the role played by the thermal-hydraulic variables, new surfaces were investigated which enhanced the role of these parameters compared to the fuel parameters. One such variable, which was highly successful, was fuel energy removal, FER(t), defined by

$$\mathrm{FER}(t) = \mathrm{FE}(0) - \mathrm{FE}(t).$$

Surfaces were generated for FER at 10 and 20 s. The four most important variables influencing FER were film boiling heat transfer, two-phase friction, subcooled discharge and slip, although not necessarily in that order. Two phase pump degradation, gap width and fuel thermal conductivity were of secondary importance, but this result is not clearly established. Response surface predictions were poorer than for PCT, OD or FE. Furthermore, 153 calculations were required to determine the 4 or 5 most important parameters (30-40 points per variable). Statistical analysis would imply that FER was a more complex surface to model, with strongly non-linear behavior suspected.

Additional surfaces were investigated based on measures of core flow. Only time integrals of core flow showed any promise of being able to provide information on core damage. Although many response surfaces were created, only two were analyzed in depth: integrated core momentum at 10 and 20 s. These results were quite typical of this class of dependent variables. Predictability of the models was very poor. Only two variables, friction and slip, appeared in all eight models for both times. The behavior of some of the independent variables was sufficiently strange that the adequacy of the statistical base could be questioned. The 153 data points are probably just barely sufficient to determine the 3 or 4 most important variables ($\sim$ 50 points per variable).

In summary, several new dependent variables have been proposed and investigated. These new variables can definitely enhance our understanding of the physical processes controlling plant behavior for many types of accidents. They should also provide important new criteria for licensing evaluations. In addition, we have expanded our knowledge of the number of calculations required to produce adequate statistical pictures of the LOCA behavior. It is apparent that the importance of fuel parameters <u>can</u> be suppressed. This procedure, however, discloses the more complex underlying thermal-hydraulic behavior, and indicates the need for larger data bases.

TABLE I
RELATIVE IMPORTANCE OF INPUT VARIABLES TO PCT SURFACE FOR CG-11 MODEL

VARIABLE	°F/σ		°F/1%	
	AT X^-	AT X^+	AT .99 N	AT 1.01 N
3 SLIP	15	-15	0.6*	-0.2*
4 FRICTION	-25	16	-0.9	0.9
6 CB-HT	22	-27	1.0	-1.0
12 POWER	-16	15	-7.6	7.7
18 PF	-77	37	-5.5	5.2
19 1/K	-26	59	-3.1	3.1
20 GAP	-83	98	-3.3	3.3

N ≡ MIDRANGE

*A + 1 σ CHANGE OF SLIP YIELDS A 67% CHANGE IN DV

A - 1 σ CHANGE OF SLIP YIELDS A 33% CHANGE IN DV

TABLE II
METAL WATER REACTION - "STAR POINTS"

T_{HIGH}	T_{LOW}	SENSITIVITY - °F/σ
1850	1857	- 2°/σ
1878	1852	9°/σ
1890	1883	2°/σ
2151	2077	12°/σ
2151	2105	15°/σ
2267	2185	27°/σ

Table III
Sensitivities of PCT Distribution to Changes in Input Means*

Variable		°F/σ_U**			°F/% Nominal		
		ΔPCT_M	ΔPCT_{90}	ΔPCT_{99}	ΔPCT_M	ΔPCT_{90}	ΔPCT_{99}
3	Slip	-15	-17	-17	-.2	-.3	-.3
4	Friction	15	17	19	0.9	1.0	1.1
6	CB-HT	-26	-28	-28	-1.0	-1.1	-1.1
12	Power	15	15	17	7.7	7.7	8.5
18	PF	35	34	35	7.0	6.6	7.0
19	K	-26	-22	-22	-2.8	-2.4	-2.4
20	Gap	48	47	38	2.7	2.6	2.2

*All based on CG-11 Model

**σ_U ≡ 1/3 upper 1/2 range

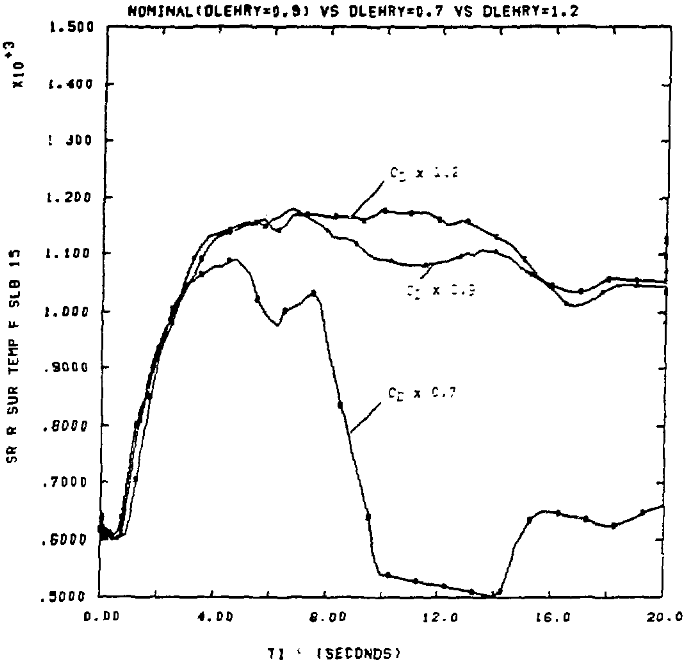

Figure 1. Effect of Subcooled Discharge Coefficient on PCT

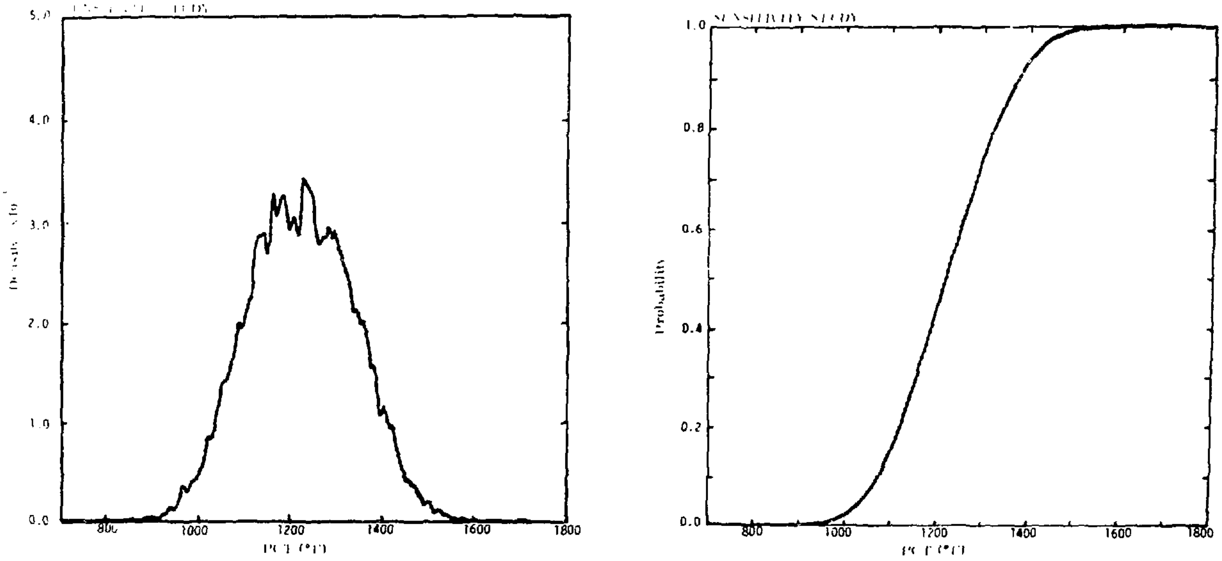

Figure 2 Plots of the Estimated Density Function and Estimated Cumulative Distribution Function for PCT Response Surface CG-11 as Determined by 10 000 Samples. (The input distributions all have means equal to nominal and standard deviations equal to 1/6 the range.)

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PREDICTION OF CRITICAL HEAT FLUX DURING TRANSIENTS*

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ABSTRACT

The onset of CHF during flow and pressure transients was predicted using steady-state CHF correlations together with a simple thermal-hydraulic code. Transient data were collected in various power-profile test sections at the ANL Freon-11 loop. The CISE Freon correlation which has successfully correlated the ANL steady-state CHF data, was employed in the transient predictions. For the slow flow-decay transients, both local-conditions (LC) and boiling-length (BL) predictions were found to be adequate. However for the rapid transients with inlet-flow blockage, the LC prediction gave better results than the BL method. Combined inlet-flow blockage and depressurization data were also in good agreement with the LC prediction. The present prediction also yielded good results for the Moxon-Edward and Cumo flow decay data. Finally a Combustion Engineering single-tube blowdown test was analyzed and all three round-tube water correlations (Bowring, Biasi, and CISE) were demonstrated to correlate the data reasonably well.

INTRODUCTION

The onset of critical heat flux (CHF) during off-normal conditions and hypothetical accidents of an operating nuclear reactor is of primary concern in the safety analyses and licensing calculations. The literature survey in Ref. 1 has revealed many experimental investigations particularly in relating to loss-of-coolant accident (LOCA). However analyses of these experiments have been very few and conclusions sometimes contradict. The present study intends to provide additional analyses for a wide range of flow and pressure transients.

ANALYTICAL DEVELOPMENT

The approach taken in the present study is (1) to calculate the fluid behavior in the heated channel during a particular transient, and (2) to predict the onset of CHF using some well-known steady-state CHF correlations. To accomplish this task, a thermal-hydraulic code here named CODA was developed. CODA is simply a one-dimensional code based on the homogeneous equilibrium formulation in two-phase regime. For a

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*Work performed under the auspices of U.S. Nuclear Regulatory Commission.
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constant area flow channel, CODA numerically integrates the following mass and energy conservation equations,

$$\frac{\partial \rho}{\partial t} + \frac{\partial G}{\partial z} = 0 \qquad (1)$$

$$\frac{\partial h}{\partial t} + \frac{G}{\rho}\left(\frac{\partial h}{\partial z}\right) = \frac{\phi P_h}{\rho A} + \frac{1}{\rho}\left(\frac{dP}{dt}\right) \qquad (2)$$

By specifying the system pressure and inlet conditions, (i.e., mass flowrate and enthalpy), these two equations together with the equation of state $\rho(h)$ can be used to solve for local-fluid parameters. One underlying assumption in this approach is that the thermodynamic properties can be evaluated at the system pressure which is taken to be constant spatially throughout the channel. Comparisons with other codes such as COBRA-IV,[2] SCORE-EVET,[3] and MECA[4] have demonstrated good agreement.[5]

In general there are two types of CHF correlations developed from steady-state data. In the first type, the local CHF is assumed to be a unique function of local quality, mass velocity, pressure, and geometry, i.e.,

$$\phi_{CHF} = \phi(x,G,P,D) \qquad (3)$$

This is the so-called local-condition (LC) hypothesis. In the second type, the CHF condition is a hydrodynamic phenomenon governed by the boiling length, mass velocity, pressure, and geometry,

$$x_c = x(L_B,G,P,D) \qquad (4)$$

This is often referred to as the hydrodynamic-condition or boiling-length (BL) hypothesis. In essence this hypothesis takes into account the upstream or integral effect. It should be noted that the BL hypothesis is mainly applicable in dryout type situations where the dryout quality and the boiling length are both positive. For steady-state data, these two approaches differ mainly with nonuniform heating. A graphical representation of these two hypotheses is shown in Fig. 1.

PREDICTION RESULTS

First some flow transients are considered. Figure 2 indicates that the LC prediction using the CISE Freon CHF correlation[6] tends to be slightly conservative in time-to-CHF for the ANL 4%/s linear flow decay experiments which were conducted in a uniformly heated tube using Freon-11 as the working fluid. The CISE correlation has been found to successfully correlate the steady-state CHF data to within ±10%.[7] The BL prediction on the other hand exhibits closer agreement as shown in Fig. 3. Here the instantaneous boiling length was used rather than the actual boiling length (or saturated two-phase path) traveled by the particle. Italian researchers have advocated the use of the instantaneous boiling length in flow transient since the dryout quality is only a slow varying function of L_B at long boiling length which is typical of these flow decay transients with dryout occurring first near the outlet.

The next set of tests consists of five rapid flow-decay experiments conducted in an outlet-peaking test section as shown in Fig. 4. During these tests, the inlet flow decayed rapidly to zero in less than 0.5 s and the system depressurized only moderately during the transients. Experimentally, CHF was observed first in the next to the last zone and this was well predicted by the LC hypothesis to within ±0.3 s as illustrated in Table 1. The BL hypothesis on the other hand predicted CHF to occur first at the outlet and then propagate upstream, this is in variance with the data. Hence, the LC prediction appears to correlate the data better and its results are summarized in Fig. 5. Subsequently, CHF prediction during transient was based on the LC hypothesis.

Moxon and Edward[9] conducted 12 tests in an uniformly heated tube with an inlet exponential flow decay which can be represented by the following equation,

$$G = 786 + 1926\ e^{-3.44t}\ \mathrm{kg/m^2s}$$

The pressure was held constant during these tests and CHF was reported to occur within the first second. This rapid CHF was primarily a result of the very high initial heat flux employed. Prediction results are shown in Fig. 6 where three round-tube water CHF correlations have been used in the prediction. Best prediction is obtained using the Bowring correlation;[10] the Biasi correlation[11] tends to overestimate the time-to-CHF slightly whereas the CISE correlation[12] yields the opposite trend. The prediction performed by Whalley using the Harwell annular flow dryout model[13] is also shown in Fig. 6; its result is very close to the CISE prediction in that both predicted CHF too early.

Cumo et al. conducted about 150 exponential flow decay tests at equivalent BWR and PWR pressures in a Freon-12 system.[14] Steady-state CHF data were demonstrated to be well-correlated by the CISE Freon correlation also, and consequently this correlation was used to predict CHF onset during these transients. Only the first 10 tests reported have been analyzed and the prediction results are shown in Fig. 7. In general, very good agreement is obtained for dryout times ranging from 0.8 to 12.0 s.

Some combined flow and pressure transient experiments were conducted at the ANL blowdown facility. The rapid depressurization of the heated section was accompanied by an inlet-flow stoppage under constant power input. Tests 170 and 315 were conducted in test sections with a symmetric-stepped profile and an inlet-peaking profile, respectively. In both tests, CHF was observed first near the outlet and then propagated upstream as shown in Figs. 8 and 9. The LC hypothesis using the CISE Freon correlation is in good agreement with the data. The curve of x = 1.0 (i.e., all vapor state) is seen to correlate the trend of these data well, thus indicative of a dryout type phenomena.

Finally, a CE blowdown experiment with flow reversal is considered. Test ST022 was a double-ended break simulation conducted in a uniformly heated tube.[15] The inlet mass flowrate was measured experimentally using the combination of densitometer and turbine flowmeter. This measurement was employed as an inlet flow-driven boundary input in CODA. CHF prediction during blowdown is shown in Fig. 10. The Biasi correlation is seen to predict CHF onset well but underestimates the CHF region whereas the CISE correlation is able to predict the region in CHF but slightly underpredict the time. It is important to note that no CHF was predicted during the early flow reversal period and in this test CHF was a direct result of liquid depletion in the heated tube.

SUMMARY

In summary, the present analysis using a simple thermal-hydraulic code in combination with steady-state CHF correlation is able to predict CHF onset during a wide range of flow and pressure transients. In particular the instantaneous local-condition hypothesis is found to be adequate in these transients. Future analyses will be extended to large-scale experiments conducted in rod-bundle configurations.

NOMENCLATURE

A	flow area
D	diameter
G	mass velocity
h	enthalpy
L	length
P	pressure, perimeter with subscript
x	quality
z	axial distance
ρ	density
ϕ	heat flux
Subscripts	
B	boiling
c	critical
h	heated
sc	subcooled

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TABLE 1 . Comparison of Time to CHF for Flow Transients in an Outlet-peaking Test Section

	Zone C			Zone D			Zone E		
z, m	2.36			2.57			2.67		
$\varphi/\bar{\varphi}$	1.35			1.16			0.81		
	t_{CHF}, s								
Run ID	EXP	LC	BL	EXP	LC	BL	EXP	LC	BL
4231	4.3	4.0	4.6	3.7	3.6	4.0	4.6	3.9	3.9
4232	3.5	3.3	3.7	2.8	3.0	3.4	3.9	3.2	3.2
4241	4.9	4.9	>5.0	4.2	4.3	4.8	5.1	4.4	4.6
4242	3.4	3.8	>4.0	2.3	2.6	3.4	2.6	2.8	3.2
4243	1.9	1.8	2.6	1.4	1.2	1.6	1.8	1.5	1.5

Note: $\varphi/\bar{\varphi}$ = local heat flux to average heat flux; EXP = experimental measurement; LC = local-condition prediction; and BL = boiling-length prediction.

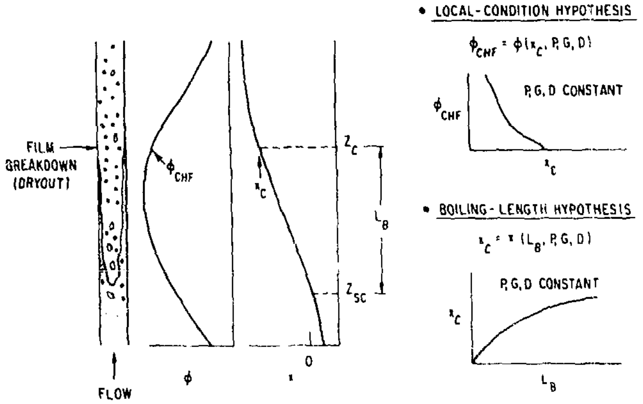

Fig. 1. Two hypotheses of critical heat flux.

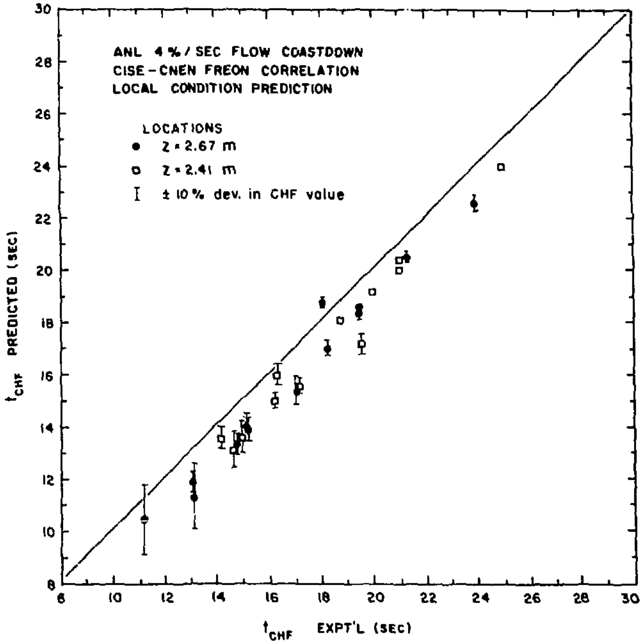

Fig. 2. Local condition prediction of ANL 4%/s flow decay transients.

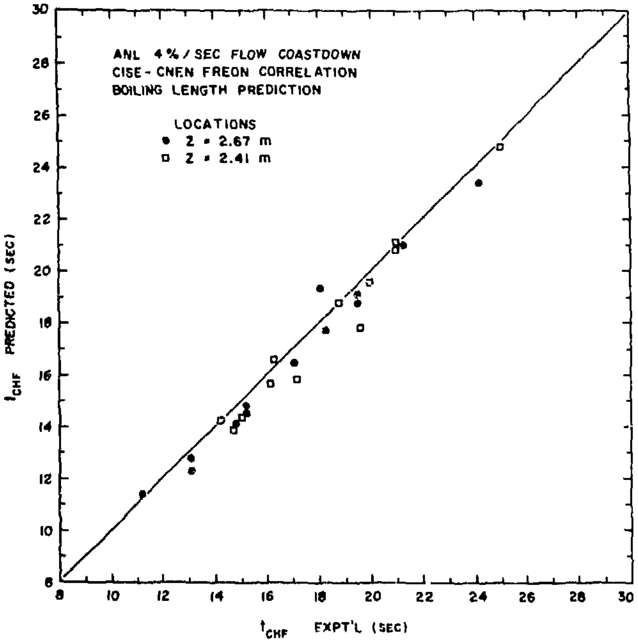

Fig. 3. Boiling length prediction of ANL 4%/s flow decay transients.

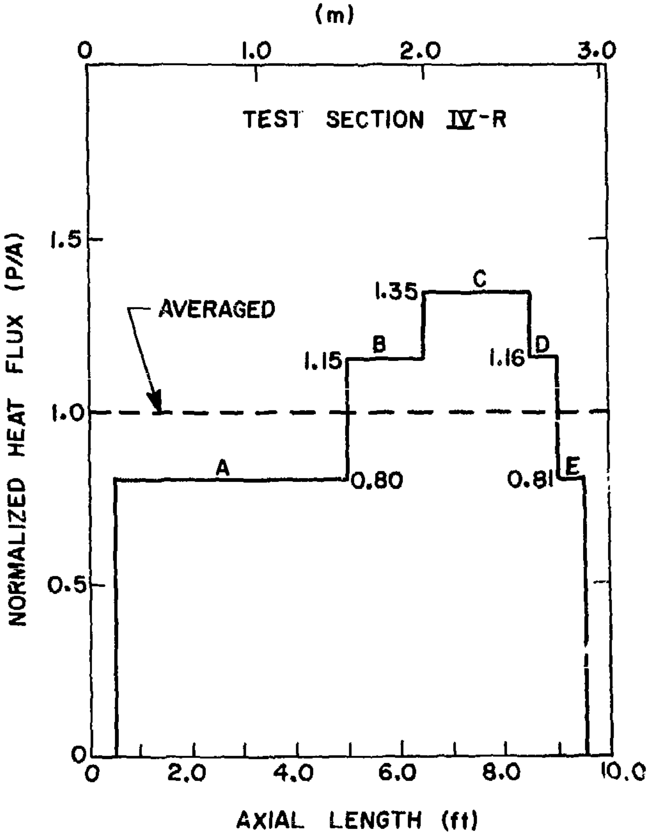

Fig. 4. An outlet peaking test section power profile.

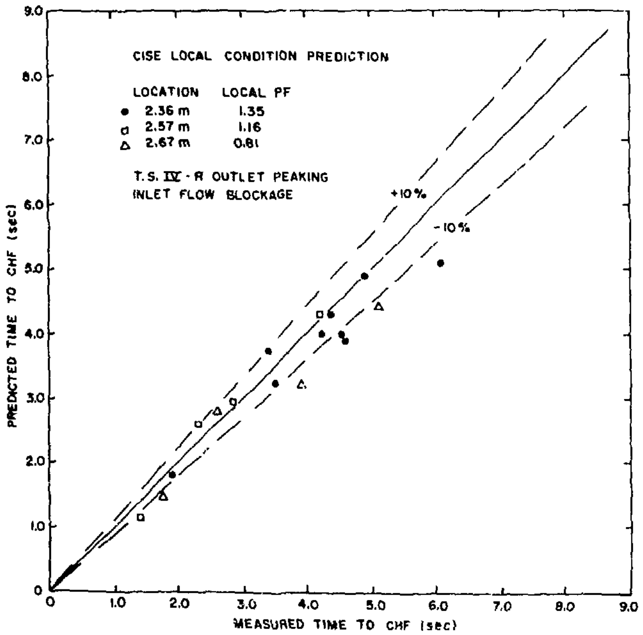

Fig. 5. Local condition prediction of ANL inlet flow blockage transients.

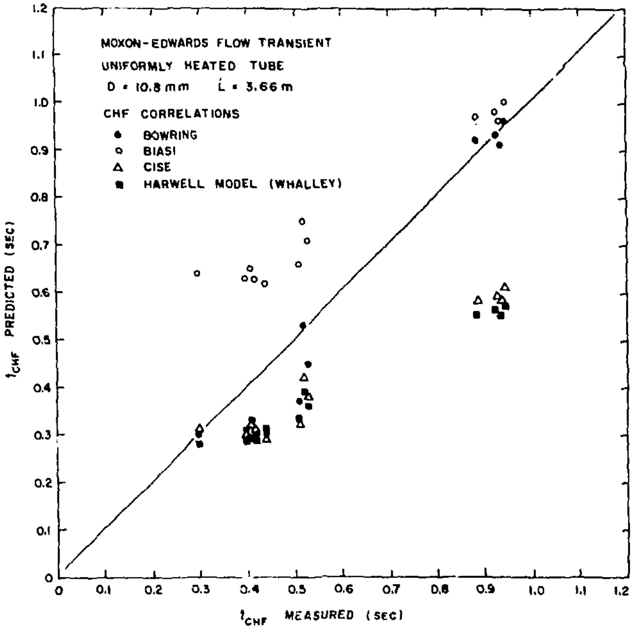

Fig. 6. CHF predictions of Moxon-Edward exponential flow decay transients.

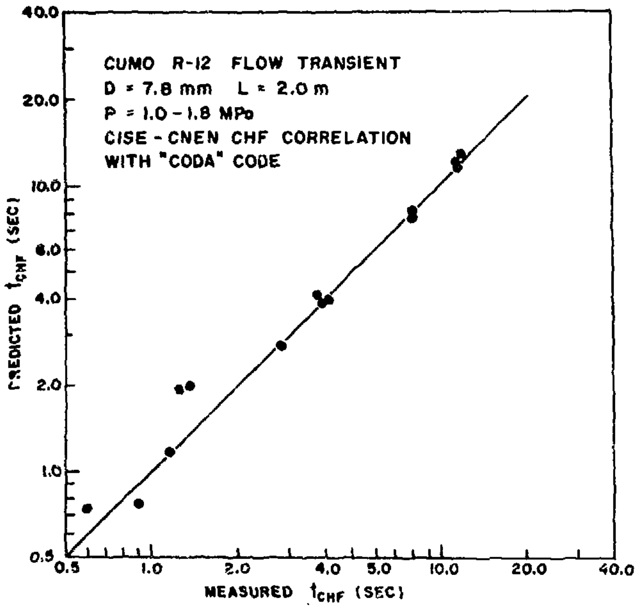

Fig. 7. CHF predictions of Cumo flow decay transients.

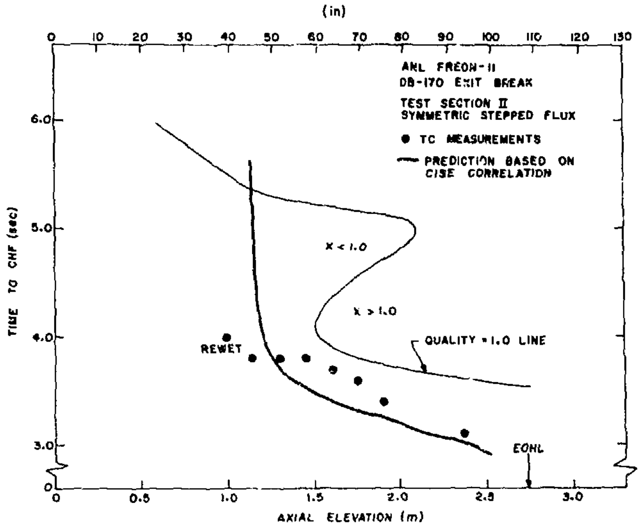

Fig. 8. CHF prediction of ANL flow and pressure transient Test 170.

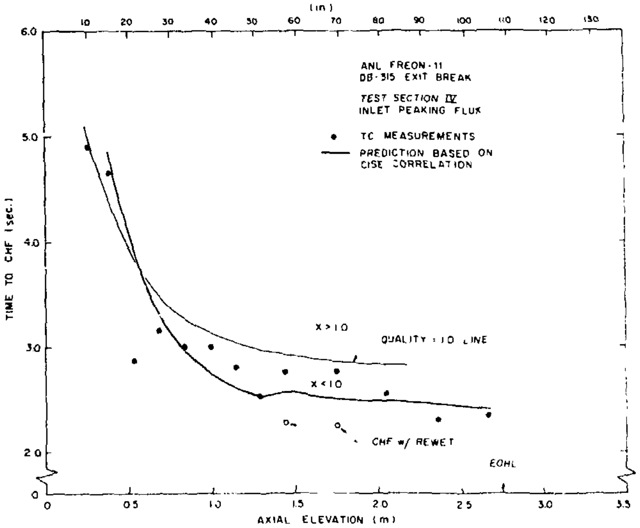

Fig. 9. CHF prediction of ANL flow and pressure transient Test 315.

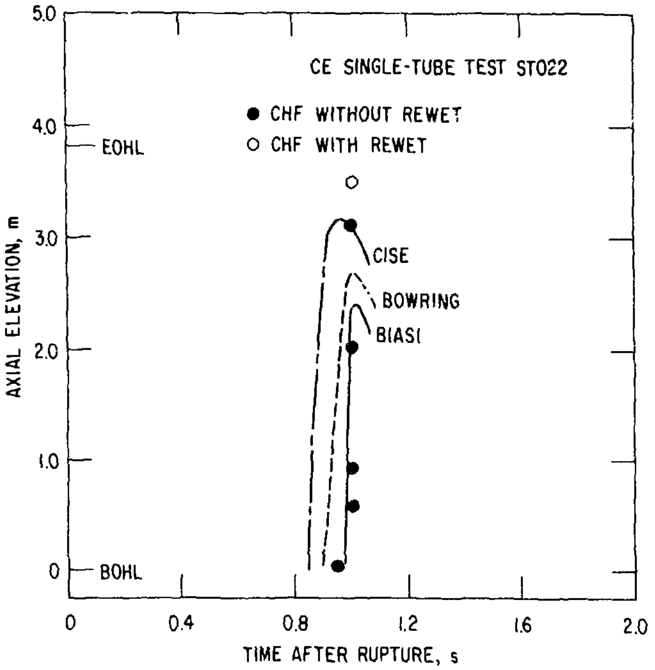

Fig. 10. CHF predictions of CE single-tube Test ST022.

STEAM GENERATOR TUBE INTEGRITY

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ABSTRACT

The integrity of the steam generator tubes is a mandatory requirement for safe operation of a pressurized water reactor. It can be shown that tubes can withstand the loading resulting from normal operation as well as from anticipated pipe break accidents even if the tubes are locally degraded by up to 70 % of the wall thickness. The eddy current technique applied allows the detection of wall thinning of less than 20 %. The operating performance of KWU steam generators shows that the corrosion rate, if any, is so small that an unacceptable wall degradation cannot occur between two inspections of the steam generators.

INTRODUCTION

The tubes in steam generators of nuclear power plants with pressurized water reactors constitute the link between the radiologically active primary circuit and the inactive steam water circuit. The integrity and leak tightness of these tubes is therefore a mandatory requirement for safe operation and high availability of the plant.

This paper describes the safety margin of KWU steam generator tubes against a leak during normal operation as well as against a tube failure during an anticipated loss of coolant accident.

DESIGN AND OPERATING CONDITIONS

The tubes of KWU steam generators are made out of Incoloy 800. The outer diameter is 22 mm; the wall thickness is 1.2 mm. The design condition of the tubes is defined at 100 % power, by the difference between the maximum possible pressure on the primary side, which is 176 bar and the minimum live steam pressure on the secondary side, which is 68.7 bar. The pressure difference of 107.3 bar results in a hoop stress of 100 N/mm^2 which is about 50 % smaller than the maximum allowable value of 190 N/mm^2. This means that a tube with only 0.61 mm wall thickness, equal to 50 % of the actual wall thickness, would meet the requirements of the specification. The pressure difference during normal operation is smaller and amounts to 89.3 bar. The safety factor against a failure during operation is thus 6.8 and as a consequence, more than twice as high as required by the specification. During normal operating conditions, both primary membrane stresses and secondary stresses are considered.

These stresses vary with the location in the steam generator. The different thermal elongations of the hot legs and cold legs of the U-tubes cause bending stresses in the U-bends, as does tube ovality in the bends. Bending stresses must also be considered because of the different thermal elongation of the economizer baffles. However, bending stresses because of tube vibration are negligibly small. The superposition of all these stresses shows that the resulting stress intensity of 195 N/mm^2 in the U-bend region is only about 30 % of the allowable value of 569 N/mm^2. In the economizer region the stress intensity of 260 N/mm^2 is also far below the allowable value. Thus locally degraded tubes meet the requirements of the code up to a wall thinning of 70 %. Because of the very conservative design of the tubes, the integrity of the tubing can be demonstrated for anticipated loss of coolant accidents, even if considerable wall degradation by corrosion or by any other mechanism is assumed.

In the event of a steam line break the pressure difference across the tube wall can reach the full primary pressure of 157 bar. The increased mass flow across the U-bends induces additional stresses which amount however to only 2 N/mm^2 and can therefore be neglected. Burst tests under operating temperatures with artificially degraded tubes showed that tubes can withstand the forces from a steam line break without exceeding the limits of the code even if the tube wall degradation is as much as 80 %. The dynamic bending and collapse loads resulting from a hypothesized main coolant pipe break can be borne by the tubes even if the tube wall is degraded by as much as 75 % of the wall thickness. This was demonstrated through a series of collapse tests with tubes having artificial defects and artificial ovality.

Fig. 1 shows the large safety margin in the tube wall thickness. 1.2 mm is the actual wall thickness; 0.61 mm is required by the specification, but only 0.3 mm are necessary at locations of defects in order to withstand the forces of pipe break accidents.

For safe operation of the plant it is mandatory that the safety margin in wall thickness not becomes less without detection during fabrication of the steam generators or during operation of the plant. The standard of quality control during tube fabrication is as high as that applied during fabrication of the pressure boundary of the primary system. That means that every single tube undergoes all specified tests. In addition the tubes are installed under extreme clean conditions in order to avoid even the slightest damage to them. Tube-to-tubesheet welding is performed by using welding equipment which eliminates the influence of the welder and guarantees the highest possible quality of the welds.

The KWU steam generator concept, as defined by the steam generator design, the selection of tube material and the water chemistry, excludes any unexpected thinning of the tube walls. The tube material Incoloy 800 is most resistant to any type of localized corrosion in the environment of a steam generator. The low phosphate treatment of the steam generator water with 2 to 6 ppm PO_4 maintains the high pH-value necessary in the steam generator to minimize corrosion attack. However, a disadvantage of phosphate is wastage attack in localized areas where concentrating mechanisms can operate.

In view of this effect, the design of the steam generators was very thoroughly investigated for the presence of possible flow stagnation zones. The present design eliminates zones with low mass flow and therefore minimizes the potential for crud deposition and for concentration of impurities. This was achieved by the installation of flow distribution baffles above the tubesheet as well as by eggcrate type tube spacers. The results of numerous flow distribution tests and corrosion tests in refreshing autoclaves and in model steam generators have proven the adequacy of this concept.

OPERATING PERFORMANCE

In order to check the tube integrity of operating steam generators hydrostatic pressure tests and eddy current tests of the tubing are performed periodically. Eddy current inspections of the steam generator tubing are performed every four years by means of automatically operated devices. The inspections are staggered, such that every year one of the four steam generators of a four loop plant is tested. Since the four steam generators of a plant are identical in design the condition of the tubes is,for all practical purposes, checked every year. The present state of the art of eddy current testing allows the detection of wall degradations of less than 20 % of the wall thickness, even at tube spacer locations. Thus tube defects can be detected which are much smaller than the allowable maximum defect depth of 70 %. Operating experience with KWU steam generators shows that the corrosion rate, if any, is so small that critical wall degradation cannot occur in the period between two inspections.

At present 30 steam generators supplied by KWU are in operation (Fig. 2). The cumulative operating time so far is about 160 years. 29 eddy current inspections have been performed to date, in each of which about 1500 tubes were inspected. In addition a total of 17 tubes has been removed from five plants in order to confirm the results by metallographic examination of the eddy current inspections.

Except for the Obrigheim steam generators no steam generator has suffered tube leakage. The two Obrigheim steam generators are the only ones tubed with Inconel 600 and have been operated since 1969 on AVT. The metallographic examination of five removed tubes revealed intergranular stress corrosion attack on the outer surface of some tubes just above the tubesheet and on the inner surface of every tube.

The four Stade steam generators are nearly identical to those of Obrigheim. The only difference lies in the tube material, which is Incoloy 800, and in the water chemistry, which has been that of low phosphate treatment since 1972. Several eddy current inspections of some 6500 tubes have revealed only crud deposits on the tubesheets but no unacceptable wall degradation. The metallographic examination of five tubes removed from the area of deposits showed no sign of stress corrosion or pitting. Only a slight wall thinning over a length of 20 mm was found at the location of the deposits on the hot side of the tubesheet. Eddy current inspections of the Borssele and Biblis B steam generators have shown some indications of wall thinning in the sludge area above the tubesheet on the hot leg. Comparison of the eddy current signals of succeeding inspections suggests a corrosion rate for the affected tubes which is so small that unacceptable wall thinning cannot occur within the period between two inspections. None of all these steam generators has flow distribution plates. The first flow distribution plates were installed in the steam generators of Gösgen, which began operation in early 1979. An eddy current inspection scheduled for June/July of this year will prove effectiveness of such plates in minimizing or eliminating crud deposition on the tubesheets.

The denting problem cannot occur in KWU steam generators because the flexible tube support system does not allow concentration of impurities. In addition the stabilized stainless steel material used is not subject to fast linear growth of magnetite.

CONCLUSIONS

- An allowable degradation of 70 % of the tube wall thickness will still enable the steam generator tubes to withstand all forces from normal operation as well as from pipe break accidents.

- The eddy current inspection technique can locate wall degradations of less than 20 %.

- The extensive quality control measures employed guarantee that the tubes will not be damaged during fabrication.

- The performance of operating plants shows that the corrosion rate, if any, is very small.

Accordingly it is concluded that undetected degradation of the tube wall thickness cannot occur and that failure of the tubes during normal operation or during a hypothesized pipe break accident can be excluded.

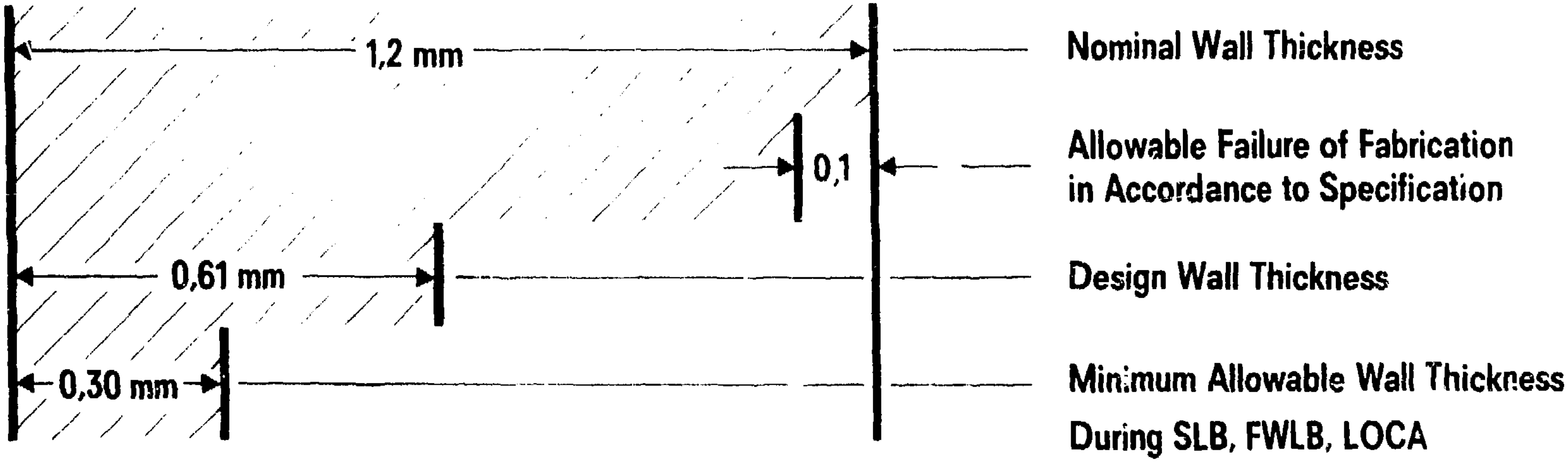
1,2 mm
Nominal Wall Thickness
0,1
Allowable Failure of Fabrication in Accordance to Specification
0,61 mm
Design Wall Thickness
0,30 mm
Minimum Allowable Wall Thickness During SLB, FWLB, LOCA

Fig.1

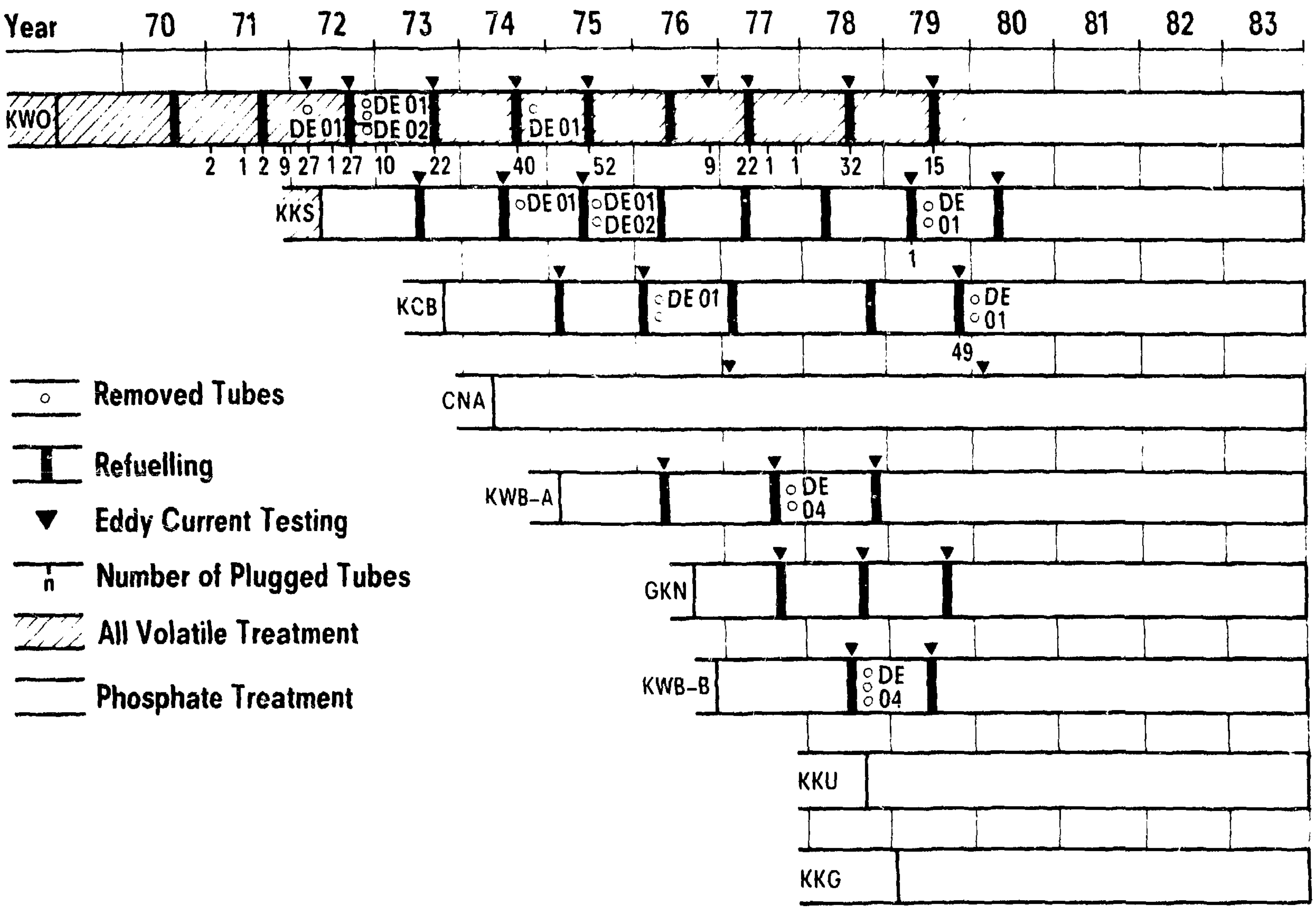

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Fig.2

UNRESOLVED PROBLEMS RELATED TO THE PERFORMANCE OF CONTAINMENT RECIRCULATION SUMPS

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ABSTRACT

Experiments are being conducted at full geometric scale to investigate the performance of containment sumps during the recirculation mode of the ECCS. Such sumps are prone to vortex formation and perform poorly if break-flow jets impinge nearby. If they do not function well, air and poor flow conditions can exist in the pump approach piping. Since these systems must function for long periods of time unattended, it is necessary to understand the effect of flow configurations, geometric configurations, and vortex suppression techniques.

INTRODUCTION

In the event of a loss of coolant accident (LOCA) in a nuclear power station, the emergency core cooling system (ECCS) and containment spray systems (CSS) would be activated to mitigate the consequences. The ECCS supplies coolant to the reactor core and vessel to dissipate the decay heat and also supplies coolant to the CSS to reduce containment pressure and scrub radioactive material from the containment environment. At first, these systems draw water from a large supply tank. Later, they are switched to recirculate water from that which has accumulated in containment. The systems are expected to operate for extended periods of time in this mode.

Sumps are provided in the containment to collect water and supply it to the ECCS pumps. They function to screen out debris and provide sufficient suction head for the pumps. The containment recirculation sumps are, thus, a key flow link in providing coolant to the reactor and in providing control of the containment environment during recirculation mode.

The character of the flow in lines leading to safety system pumps is, to a great extent, determined by phenomena occurring in the containment sump. Of primary concern is the tendency for vortices to form either because of the design approach and sump geometry or because of artificial initiation by, say, debris blockage of screens. Additionally, direct impingement of jets in the sump area, as might be caused by a nearby pipe break, has recently been identified as contributing to poor sump performance. Figure 1 shows a free surface vortex and a simulated break-flow impingement jet near an inlet.

Flow conditions in containment sumps could adversely affect the performance of these safety systems. As the licensing authority, the Nuclear Regulatory Commission (NRC) is concerned about the reliability of the safety systems in the recirculation mode. Specifically, the major items of concern are:

1. Entrained Air - Air entrainment in the suction lines could be due to air entraining vortices existing in the sump, or due to air entrainment generated by water or steam jets from the break impinging near the sump through the free water surface. It has been established that even a low air concentration in a suction line, such as 3 to 5%, can lower the efficiency of pumps considerably. In fact, certain centrifugal pumps are known to lose prime when the air concentration by volume exceeds about 15% of water volume.

2. Prerotation - The various possible approach flow patterns, together with possible screen blockages in the sump, could induce a swirling flow in the sump area. This swirl could be transmitted to the suction pipe and might increase the losses at the intake. Prerotation could also affect the performance of pumps located close to the pump.

3. Losses Leading to Insufficient NPSH - A poorly designed sump could result in excessive head losses. Entrance losses caused by swirling flow, the pipe inlet geometry, and vortex suppressors, may add up to a value such that the required NPSH of the pump is not satisfied. It may be noted that the water temperature also affects the NPSH due to changes in vapor pressure.

Historically, experiments with models of sumps and inlets have been used to guide designers in their specification of geometry. Such studies, typically conducted at scales of the order of 1:50, require the application of some extrapolation criterion because the two primary scaling parameters, Reynolds and Froude numbers, cannot simultaneously be achieved in a model. There will thus be a difference in behavior of model and prototype, scale effect, which the extrapolation technique or experience must bridge[1]. This aspect of scale model testing is presently the subject of considerable research and not presently settled. In many engineering applications, some ambiguity in predicting prototype performance can be tolerated inasmuch as feedback between operating prototype performance and model studies has a long history. In addition, less than optimum performance is often tolerated and can often be remedied in the field.

In the experimental evaluation of containment sumps, however, it is not possible to test the prototype under fully simulated accident conditions. It is often not possible to use the full containment floor, but only a blocked-off portion so that approach flow conditions are not fully represented. Further, testing at elevated temperature with various blockages and impinging jets is not feasible. The trend, then, has been to conduct model tests at scales of the order of 1:3 in order to minimize scale effects at reasonable cost. While this is a good approach to conventional sumps, additional complications are often introduced in containment sumps. In particular, various types of vortex suppression devices are often employed and the associated scale effects have not been extensively studied. In order to remove the effect of scale and concentrate on the parameters which directly affect sump performance and vortex suppression, the present study is being conducted at full geometric scale.

BACKGROUND

Since the formation of vortices at intakes is a function of Weber number, Reynolds number, Froude number, and geometry, it is of primary importance to fix or eliminate at least three of the variables in order to establish a cause/effect relationship in vortex formation. The purpose of the proposed work is to conduct studies at a large enough scale (approximately 1:1) that the Weber number effects are insignificant while the Froude and Reynolds numbers are essentially those of the prototypes. This leaves the functional relationship

$$\text{vortex formation} = f\ (\text{geometry})$$

as the primary dependence for vortex formation. Measures of the vortex formation tendency include observed vortex type, pipeline swirl, inlet loss coefficient, and pipeline air content.

The flow situation that could initiate a vortex in a containment recirculation sump could be very complex, as the contributing sources include flow distribution between possible sources, the approach geometry, flow rotation generated by partially blocked screens and gratings, and rotational wakes generated by obstructing objects such as columns and peripheral equipment[1]. Tests at ARL have shown that the values of critical submergence needed for vortex-free operation commonly found in the literature on pump intakes were inadequate in the case of recirculation sumps.

Both free surface and submerged vortices (which terminate on the floor or walls) have been found in containment sump models. Often the flow field is dominated by screen blockage, which gives rise to offset flow introduction which is a natural rotation source, and by break flow impingement in the vicinity of the sump. The former usually leads to strong free surface vortices so that designers often specify that vortex suppression devices be fitted. The latter leads to significant air bubble entrainment for which the only remedy available is to provide a longer flow path (timewise) to allow the air to escape to the free surface.

METHOD OF INVESTIGATION

The objectives of the present investigations are (i) to identify the interrelationships and relative importance of the multitude of flow and geometric parameters on the hydraulic performance of containment recirculation sumps, and (ii) to examine the effectiveness of various vortex suppression techniques in situations with worst geometric and flow parameter combinations. To avoid questions regarding the extrapolation of results from model to prototype, the facility is essentially full scale. Any of the parameters of concern can be varied within definite ranges so that the functional relationship between vortex formation and geometry can be mapped. In addition, the available flowrate and submergences cover a wide range while break and drain flows can be simulated. Once the functional relationship has been established, selected configurations will be utilized to evaluate existing and proposed methods of vortex suppression.

The performance of containment sumps is characterized by the nature of the flow in the suction lines of ECCS pumps. Approach flow patterns, vortex formation, and impingement of break and drain flows all contribute to swirling flow and entrained air in the lines as well as affecting the composite loss coefficient[2].

The observed free surface vortices are a good indication of sump performance and a numerical scale is used which is indicative of the types which form. The graduations run from "0" for no visible activity to "6" for a vortex with defined air core entering the inlet. Intermediate numerical values are assigned to discernible stages of development. An observer enters the vortex type on a keypad at preselected intervals. These data are then available for time series analysis in the acquisition system. Further documentation of the observations can be achieved using photographs, movies, and video recordings.

Pipeline swirl is indicated by crossed-vane swirl meters commonly called vortimeters. These devices rotate about the pipe central axis and the vanes span about 75% of the cross-section. Under most circumstances, the angular rotation speed is indicative of the average swirl angle of the rotational core region of flow[3,4].

The inlet loss coefficients are established by measuring the hydraulic gradeline in the discharge lines and extrapolating back to the line entrance[5]. Ten piezometers are provided in each line and individual locations are selected via a scanning valve under control of the data acquisition system. This system also scans differential pressure flowmeters installed in the pipe network.

The void fraction due to air transported in one discharge line is determined using a conductivity meter of the rotating electric field type. The conductivity is measured and is proportional to the conductive component of the two-phase flow.

DESCRIPTION OF THE FACILITY

The test facility was designed so that any of the flow or geometric parameters of the sump could be varied over typical ranges with least time and effort by simple alterations of floors, walls, and pipe fittings. The test setup consists of a concrete main tank, 70 ft by 35 ft by 12-1/2 ft, and a concrete sump tank, 20 ft by 15 ft by 10 ft, situated within the main tank. Inflow is distributed along three sides of the main tank and provision was made to produce non-uniform approach flows using blockage. False walls and tank floors were provided such that sump geometries could be varied. Four rows of outlet holes in the front wall were provided with each row having five holes of 25 inch diameter at 4 ft centers. Sets of two holes in a row were used to attach the suction pipes which could be of any diameter in the range of 8 inches to 24 inches.

The suction pipes extend from the sump tank to a suction chamber 50 ft away and are long enough to facilitate swirl, pressure gradient, and discharge measurements. Each of the suction pipes accommodates a vortimeter for swirl measurement and ten pressure taps, one pipe diameter apart for pressure gradient measurements. Flow in the suction pipes can be remotely regulated and measured.

Two vertical pumps, one diesel driven and another electrically driven, recirculate up to 20,000 gpm to the main tank. Up to 60% of the total flow can be delivered as breakflow and/or drain flow simulation at a higher elevation or as high velocity jets.

TEST PROGRAM

The test program is designed to have broad application and it will serve a two-fold purpose. First, the test program will provide an extensive data base to the NRC, which is presently unavailable, for the resolution of sump vortexing problems, a part of unresolved safety issue A-43. Additionally, the same data will provide ECCS sump design information to the nuclear industry in general.

The test program covers three broad areas of interest for ECCS sump design. They are:

1. The fundamental behavior of the sump with uniform approach flow conditions.
2. Changes in the fundamental behavior of the sump as a result of adverse conditions caused by flow disruptions.
3. Vortex suppression.

Figure 2 shows that the typical ECCS sump is described by at least 11 geometric variables in addition to two flow variables (velocity, U, and submergence, s), and 3 fluid properties (kinematic viscosity, ν, density, ρ, and surface tension, σ). Standard dimensional analysis using a single length dimension, the pipe diameter, d, gives

$$\text{dependent variable} = \text{function}\ \left(\frac{L}{d}, \frac{B}{d}, \frac{a}{d}, \frac{b}{d}, \frac{g}{d}, \frac{c}{d}, \frac{e_x}{d}, \frac{e_y}{d}, \frac{f}{d}, \frac{s}{d}, R, F, W\right)$$

where the dependent variable may be severity, swirl, losses, etc.

Here, R, Reynolds number = Ud/ν, F, Froude number = $U/\sqrt{Gs}$, and W, Weber number = $U^2 d/(\sigma/\rho)$.

The result is three dimensionless numbers and ten dimensionless geometric variables that are non-dinensionalized using the pipe diameter. Clearly, an ECCS sump is a multiple length problem whose behavior cannot be analyzed empirically with only a few dimensionless independent variables as many simpler sump geometries can. For this reason, statistical methods are employed. In particular, a fractional factorial experiment design is used.

The fractional factorial technique is particularly well suited to problems where there is a large number of independent variables, yet there is also some information available on the expected behavior due to some of the independent variables. The two principle advantages of using a fractional factorial method are that it will give the variable effects and some variable interactions, and it will allow each variable to be investigated over its range with a relatively small investment of resources.

The tests are divided into two categories which are fractional factorial experiments and sensitivity tests. The variables for the fractional factorial tests are selected using a judgmental ranking of each variable according to their expected importance along with certain physical constraints imposed by the test facility. The sump variables and topics of concern that remain are tested using sensitivity tests. The behavior of the parameter under investigation is assumed to be independent of the other variables, i.e., the parameters are studied one at a time while holding all other variables fixed.

For example, part of the test program is a $(1/3 \times 3^4)$ fractional factorial for the variables d, L, B, and b, which leads to twenty-seven test configurations and will include analysis of variance, regression analysis, and dimensionless empirical correlation. One result will be a quadratic expression for the sump's behavior of the form

$$Y = \alpha_o + \alpha_1 d + \alpha_2 L + \alpha_3 B + \alpha_4 b + \alpha_{12} dL + \alpha_{13} dB + \alpha_{23} LB + \alpha_{11} d^2 + \alpha_{22} L^2 + \alpha_{33} B^2$$

where Y is the dependent variable and can be vortex type, swirl, loss coefficient, or void fraction.

The sensitivity experiments test both sump parameters, sump orientation, and configuration concepts. This testing investigates the following 12 items:

1. non-symmetrical orientations
2. vertical outlet pipes
3. BWR pipe configurations
4. effect of variable e_y
5. effect of variable x
6. effect of variable c
7. cover plate issues (variable a)
8. 8 inch diameter outlet pipes
9. scale modeling effects
10. temperature effects
11. single outlet pipe operation
12. outlet pipe shape

The sensitivity tests are performed by varying one of the above sensitivity parameters through several configurations (usually three), while keeping the other geometric sump parameters fixed.

Another part of the test program investigates the behavior of the sump when subjected to approach flow perturbations - both above and below the surface. The flow disturbances considered are screen blockage (up to 75%), break flow impingement, drain flow impingement, flowrate transients, and non-uniform approach velocity distribution and obstructions.

The test procedures are similar with emphasis on establishing the inherent stability of severely perturbed sump designs. A sump excursion from the baseline behavior provides data which will determine trends in vortexing, determine the movement of any fundamental boundaries, determine increased or decreased profile losses, etc.

A sustained air core or debris entraining vortex can lead to high inlet losses, to decreased pump performance, or even to pump failure. The development and evaluation of vortex suppressors is important in many situations. Generally, sump vortices and flow rotation in a sump are suppressed using two processes: form drag and viscous dissipation. A vortex suppressor exhibits a large form drag in the direction of expected flow rotation which suppresses the sump's angular rotation. Additionally, suppressors break up large scale motion so that the resulting smaller scale motions will dissipate rapidly. Vortex suppressor configurations under investigation consist of horizontal grates, inner cages, and splitter vanes.

SUMMARY

The primary function of ECCS containment sumps is to scavenge and filter water from the containment floor during recirculation mode and provide flow free from debris, air, and swirl to the pumps with a minimum of head loss. The two major problems encountered result from the tendency of such inlets to form vortices and from the possibility that a jet due to a pipe break may impinge in the sump vicinity.

Performance testing of these sumps is presently conducted in situ or using large physical models. Comprehensive testing in the former case is not feasible for logistical reasons. Model testing is site specific and requires complex modeling procedures for extrapolation of results to the prototype.

The research program which has been described was designed to provide generic results with regard to sump flows and geometries under all possible types of perturbations to the flow. Dependent variables include vortex type, swirl, inlet loss coefficient, and void fraction. The independent variables include the flow and geometric parameters. In addition, substantial effort is being given to various vortex suppression techniques by testing under controlled conditions with baseline comparison.

ACKNOWLEDGEMENTS

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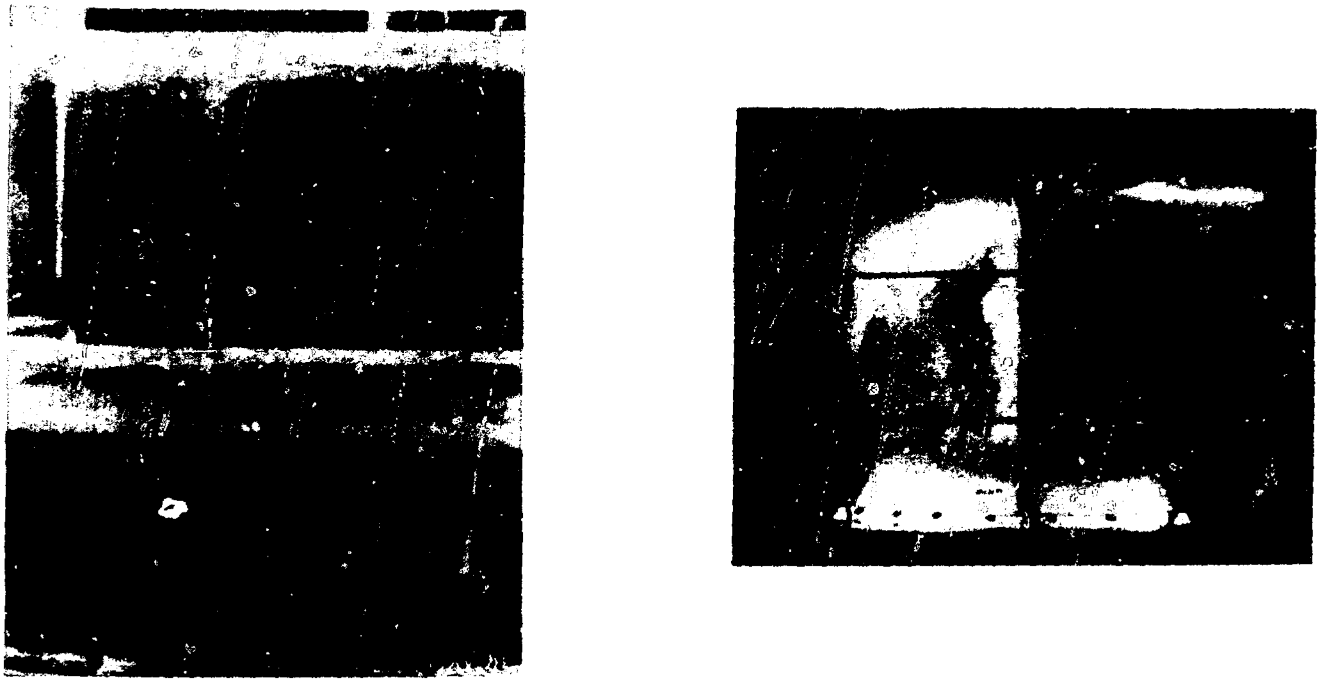
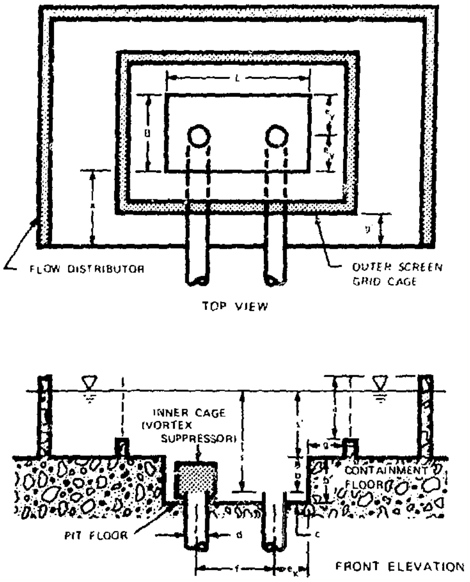


(a)

(b)

FIGURE 1 AIR-ENTRAINMENT AS OBSERVED IN SUMP MODELS
(a) DUE TO AIR-CORE VORTICES, (b) DUE TO BREAK FLOWS



GEOMETRIC PARAMETERS	TEST RANGE	GEOMETRIC PARAMETERS	TEST RANGE
s	0.5 TO 20 FT	e	0 TO 10 FT
a	1 TO 6 FT	f	4 TO 12 FT
b	1 TO 15 FT	g	0 TO 6 FT
c	0 TO 2 FT	B	4 TO 15 FT
d	8, 12, 16, 24 INCHES	L	6 TO 20 FT

FIGURE 2 GEOMETRIC PARAMETERS

PWR'S UNRESOLVED SAFETY ISSUES AS CHECKED BY DSN*

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ABSTRACT

By use of deterministic and probabilistic methods for safety analysis at the design and construction stage, several safety issues concerning French PWR plants have been identified. Some of these issues have also been identified by NRC, whereas others are specific to the French safety approach.

These issues have led to implementation of specific measures at operating plants, and are being taken into account in the design of future units.

Analysis of operating experience and incidents is also being used to verify and validate existing designs, and identify any deficiencies in design and procedures, particularly at the level of plant operation and man-reactor interfaces.

1 - INTRODUCTION

An unresolved safety issue, according to our own definition, is a safety aspect of plant design, construction or operation considered capable of improvement or better substantiation, referencing to recent technical developments, pertinent new knowledge, and plant operating experience.

Resolving such issues should substantially enhance safety and its demonstration, and may lead to modifications in plant design and operating rules, either by backfitting of plants already in service, or for future plants.

This paper points out differences between the French and US positions, and :

- shows how French safety analysis methods have been used to identify unresolved safety issues,
- discusses some of these issues.

2 - IDENTIFICATION OF UNRESOLVED SAFETY ISSUES

The general approach to safety analysis in France must be situated within the context of the French nuclear program, which is characterized by its size, and by the high degree of plants standardization (2 standard plant designs : 900 and 1300 MWe).

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900 MWe units have been designed and constructed in accordance with US regulations, plus some specifically French regulations. Design reviews and other safety-related actions have demonstrated that the safety of these plants is satisfactory, even if certain specific aspects require further studies and investigations.

This assessment of 900 MWe units were the basis for a detailed examination of fundamental PWR safety features, unresolved issues, and necessary studies, performed concurrently with design review of the first two 1300 MWe units.

Results from these procedures and from initial 900 MWe plant startup and operating experience were then used to specify the main safety requirements and features for subsequent 1300 MWe plants [1].

The rest of this paper surveys the main safety issues identified to day. They may be divided into two categories :

- issues identified by safety analysis during plant design and construction,
- issues identified by safety analysis of operating plants.

2-1. Issues identified by safety analysis during design and construction

At this step the purpose of safety analysis is to ensure compliance with the defence in depth principle, which involves applying a set of deterministic rules concerning the entire plant, its individual systems and components, and accident studies.

It has been possible to verify the consistency of some of these rules using probabilistic methods, including reliability studies performed by Electricité de France (EdF) on the main systems of the Fessenheim plant. Results of these studies were applied to subsequent plant units, this being helped by the standardization of plant designs.

Therefore, existing safety rules were modified or completed, specially according to the following ideas, which are the subject of ongoing studies :

- Plant design situations : French safety authorities have defined [1] objectives for the establishment of a new list of design situations, and have requested that the existing list (based on ANSI N.18.2) be completed by studying additional situations whose probability is non-negligible or uncertain. Among these are failure of redundant systems (cf. § 3-1.) and previously non postulated pipe breaks (cf. § 3-2.).

- Capability of equipement to operate correctly under all circumstances :

  . Design rules : Details in application of the ASME code to class 2 and 3 systems were interpreted in a different way by safety authorities and EdF. After discussion, subsequent requirements concerning equipments have been precised. These requirements have been correllated with the equipment function during accident situations (cf. § 3-3.).

  - Equipment qualification : Recent research has improved specification of post-accident environmental conditions, and this needs additional qualification of electrical and mechanical equipment (cf. § 3-4.).

- Single failure criterion : The definition of this criterion has been precised. Details of its application to PWR plants have been defined jointly by the safety authorities and EdF. The resulting problems are the subject of ongoing studies (cf. § 3-5.).

- Interactions between systems and common mode failures : These problems cannot be handled only by applying purely deterministic rules during design. It is necessary to review in detail as-constructed plants, to detect any deficiencies. This review has been started for both 900 MWe and 1300 MWe units.

- Accident studies : Accident studies performed by EdF and the NSSS vendor are mainly intended on doing design, and for this reason are based on "maximum" accident scenarios, involving very conservative assumptions. Estimations of available margins between these assumptions and actual situations are used to select and define subjects for long-term safety studies (e.g., LOCA studies).

  However, accident studies must also supply information for establishment of plant operating procedures, and for design of instrumentation for accident diagnosis and safe reactor control. At a very early stage the French safety authorities were conscious of this need and they therefore asked EdF to study post-accident situations and determine conditions to drive back the reactor to a safe shutdown condition. The importance of these studies has been recently highlighted by the TMI accident, and their results are being used to identify and resolve safety issues related to ensuring safe control of reactors under all circumstances.

2-2. Issues identified by analysis of plant operation

Analysis of equipment failures, operating incidents, accidents, and results of periodic inspections is a potentially profitable source of informations on plant safety.

o Such analysis is used to check the validity of design and construction assumptions :

  - Behaviour of reactor components is evaluated under actual operating conditions. In this respect, there are currently two outstanding concerns : primary coolant activity (cf. § 3-6.) and the integrity of steam generator tubes (cf. § 3-7.).
  - Unforeseen operating problems have been revealed, including abnormal pressure transients (cf. § 3-8.).
  - Operating experience is necessary for verification of data and assumptions used in reliability studies, and for identification of the causes of common mode failures. In this context a common failure mode may

concern the redundant components of a system, several systems, or identical equipment at a large series of virtually identical standard-design plants. For this reason, any simultaneous safety related equipment *unavailability* or *startup failure* is very carefully studied.

- Plant operation analysis also identifies events providing "advance warning" of potentially serious accidents. Identification of these events is in itself a safety issue, and the subject of ongoing studies.

France does not have enough PWR operating experience to draw up a representative list of safety problems encountered during plant operation. However, it is already clear that considerable attention must be paid to consequences of human errors, and behaviour of equipment common to a series of standardized plants.

Mention should also be made of the main lessons learned from the Three Mile Island accident, which are discussed in reference 2.

3 - CURRENT PROGRESS ON SELECTED SAFETY CONCERNS

3-1. Failure of redundant systems

Failure of frequently actuated systems has been studied at the request of safety authorities. Results of some of these studies are indicated in reference 3.

- Total loss of power (onsite and offsite) : Design of 1300 MWe units allows sufficient time for implementation of appropriate measures and restoration of a power supply. However, the corresponding procedures have not yet been defined. For 900 MWe units, this problem is still being studied. This item is similar to the issue defined by the NRC. However, the fact that plant design is standardized in France made us easier to specify corresponding safety requirements.

- Loss of ultimate heat sink : Studies by EdF have shown that reactor cooling can be ensured for several days after heat sink loss, by mean of onsite water reserves. However, corresponding procedures are not yet defined.

- Anticipated transients without scram (ATWS) : This concern is the same in the US, and has been studied by a working group. Its conclusions [3] were taken into account in the design of 1300 MWe units, which are equipped with a digital integrated safety system. Necessary modifications for 900 MWe units are under study.

- Auxiliary feedwater supply to steam generators : Reliability studies on 900 MWe and 1300 MWe reactor auxiliary feedwater systems show that the probability of system startup failure is high, taking into account the frequency of actuation. The safety authorities have therefore asked EdF to study the consequences of failure of this system, and any necessary improvements.

3-2. Non postulated pipe breaks

Application of NRC rules (SRP 3.6.1. and 3.6.2.) allows to not consider rupture of certain high energy pipes in design, provided more severe stress limits are used. The French position is that, even if all possible measures are taken to prevent rupture of a pipe, it is still necessary to verify that rupture cannot create an accident with unacceptable consequences. For this reason, protective devices have been provided for certain pipes crossing the annular space between the two containment walls at 1300 MWE units.

Another problem concerns in-core instrumentation tubes inside reactor vessels. A study by the NSSS vendor has demonstrated that an adequate core cooling is no longer possible when more than a few of these tubes are ruptured. For this reason, the safety authorities have requested use of more severe stress criteria. This item may conduct to examine the feasibility of a new design for future plants.

3-3. Requirements for safety-related equipment (ASME class 2 and 3)

Discussions have been held with EdF and the vendor on requirements related to loading combinations, stress limits and testing (interpretation of ASME code). The French position is consistent with NRC in PSRP 3.9.3.

With respect to 900 MWe reactors, verifications are underway concerning functional capability of Engineered Safety Features in the case of a LOCA, or a (LOCA + SSE) combination. For 1300 MWe reactors, these requirements are applied during the design phase. Studies are also being pursued on active component operability testing.

3-4. Qualification of electrical equipment for use under accident conditions

This issue is similar to that identified by the NRC, and concerns all plants units.

A qualification program with specification of temperature and pressure profiles, irradiation and aging conditions, earthquake-related requirements, types of test and qualification codes has been defined jointly with EdF and Framatome. This program is currently being underway for qualification of 900 and 1300 MWe plant equipment.

3-5. Application of the single failure criterion

Detailed application of this criterion to 1300 MWe PWR units has been discussed with EdF and Framatome. The concepts of active and passive failures have been more precisely defined, and safety authorities have requested that a leak at any point in a system be considered as a passive failure if the system must operate longer than 24 hours. Such a leak may, however, be considered limited in volume and duration if the portion of the system concerned can be isolated within a reasonable delay. This position is different from the NRC's one, the definition of a leak is not limited to leaks at valve and pump seals.

As a consequence, EdF was asked to study LOCA concomitant with a leakage in the safety injection system. These studies are completed for 1300 MWe units, their conclusion being that satisfactory core cooling is ensured in all cases. Studies are still on for 900 MWe reactors.

Also, due to this new definition, reactor design must now take into account partial unavailability of systems resulting from routine maintenance and testing.

3-6. Primary coolant activity

French safety authorities have always considered primary coolant activity as a safety concern, because of its impact on reactor operation and radioactivity releases.

o Reactor operation : Activation of primary coolant system materials and high-activity deposits can, in the medium term, cause considerable difficulties for plant operating staff. Moreover, the resulting increase in personnel radiation exposure can lead to lower the frequency and scope of routine maintenance and tests for equipment such as steam generator tubes.

o The impact of primary coolant on normal and accident radioactivity releases, both liquid and gaseous, is evident. To reduce the activity of primary coolant needs not only to limit radioactive sources, but also to provide instrumentation to monitor primary coolant activity and in-containment activity.

o Source limitation : French experience has allowed evaluation of the relative contributions of the sources involved (activation products, and fission products), and implementation or definition of methods for reducing total primary coolant activity, namely :
  - for activation products :
    . change of cobalt-emitting materials,
    . research on electromagnetic filtration to remove particles from the primary coolant,
    . use of a different material for fuel assembly grids,
    . modification of coolant chemical treatment : use of lithium highly enriched in lithium-7, to lower formation of tritium.

- for fission products release :
  - . analysis of the defective fuel behaviour. The fission products release contribution to total activity could become a matter of concern for plants with load follows operation.

Technical operating specifications stipulate that the plant operator must notify safety authorities as soon as primary coolant activity reaches a value corresponding to a clad failure rate of 0.03 %.

o Instrumentation : Information is needed on fuel cladding integrity before, during and after an accident and it is therefore, envisaged to implant instrumentation capable of measuring only primary coolant activity. Also, studies have been initiated on instruments to measure in-containment dose rates between 1 and 10^6 rad per hour.

During the IAEA meeting at Chalk River on September 16th to 20th, 1979, the French delegation noted that US safety authorities remain very vigilant and ferm with respect to regulations concerning levels of activity in the primary system, and that the NRC now considers itself directly concerned by problems related with clad failures during normal operating conditions.

3-7. Integrity of steam generator tubes

Connecting with the observed corrosion phenomena, the resistance of steam generator tubes under normal and accident conditions is a matter of concern for safety authorities in both France and the US.

The main French actions related to this issue are :

- inspection of steam generators at the first Fessenheim units. To day no special difficulties have been found,
- results on tube behaviour under accident conditions are now being evaluated in order to determine available margins,
- use of experience acquired from the above actions in design of future plant units, related with the importance of steam generator tubes, which are part of both the second and third containment barriers.

Finally, recent accidents involving steam generator tube ruptures need re-examining classification of steam generator tube rupture in the list of design conditions (ANSI N.18.2.).

3-8. Abnormal pressure transients

This issue has been identified by the NRC, and French reactors are concerned. Design of residual heat removal system safety valves is based on postulated accidents resulting from untimely operation of systems. However, incidents due to operator actions have not been considered in the design : for example, manual start up of a primary coolant pump when the RCS is solid.

A provisional solution involving application of suitable operating procedures has been implemented at operating plants. Studies by EdF and Framatome are underway to determine a definitive solution permitting total control of all overpressure conditions, whatever their origin. This solution will then be applied to every plants.

4 - CONCLUSIONS

- It is clear today that design-related safety issues are fairly well understood, even if they are not all resolved, and analysis of the TMI accident has not invalidated the basic design of pressurized water reactors. Study of outstanding issues may result in some modifications to the design of future plants.
- A major effort is necessary to better define issues relative to plant operation and the man-reactor interface, which are not yet totally understood. Although resulting modifications will mainly concern operating rules and technical specifications, it is nonetheless evident that some design changes will also be necessary, particularly for control rooms

ACKNOWLEDGEMENTS

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APPLICATION OF PARTIAL FAILURE ANALYSIS TO AN ACCIDENT INVOLVING LOSS OF COOLING IN A SPENT FUEL STORAGE POOL

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ABSTRACT

A method is described for extending event tree methodology to the evaluation of systems which are better represented by a spectrum of operational modes than by two discrete modes. The methods are applied to the analysis of the risk to the public from failure of the decay heat removal system in a spent fuel storage pool.

INTRODUCTION

The use of event trees in the analysis of reactor risk was an important development of WASH-1400.[1] A major drawback of event trees is the representation of the condition of a system in a binomial manner. In undertaking a program for the Nuclear Regulatory Commission to evaluate the risk to the public from Class 3 to 8 accidents,[2] a method was developed to extend the use of event trees to include a spectrum of system conditions. In this paper the method, partial failure analysis, is described and is applied to an accident in a spent fuel storage pool involving failure of the heat removal system. The accident sequence examined is of particular current interest because of the trend to high density storage of spent fuel and plans for the interim storage of fuel in remote facilities.

PARTIAL FAILURE ANALYSIS

In order to extend the binomial approximation of event trees to include the analysis of systems which are better characterized by a spectrum of operational modes, methods of partial failure analysis have been developed. A variety of approaches to partial failure analysis are possible. In general, the methods involve the following steps:

(1) Develop a simple relationship between the consequences of an accident and the variables describing the system, e.g., $C = f(x_1)g(x_2)h(x_3)$.

(2) Identify the source of uncertainty in variables as being random, in that different reactor states could be encountered depending on the

specific timing or circumstances of an accident, or as being systematic, resulting from ignorance as to the true value of the parameter.

(3) Develop probability density functions for those parameters that are random in nature.

(4) For each variable quantify the systematic uncertainty interval that exists as the result of ignorance. Typically 90 percent confidence intervals are used although in some cases subjective judgment must be applied.

(5) Develop a parametric representation of the probability density function for each variable in such a manner that the range of systematic uncertainty is spanned by varying the parameters.

(6) Using Monte Carlo analysis randomly select a set of parameters that characterize all of the density functions. From these density functions, develop a complementary cumulative distribution function for accident consequences. If the density functions and the relationship between the variables and the consequences are sufficiently simple, this can be done analytically. In general, the CCDF can be obtained by Monte Carlo analysis.

(7) Perform a large number of calculations in the manner described in the previous step.

The shape and centrality of each of the CCDF's obtained in step 6 above, are defined by the variables which exhibit random variation. From the group of CCDF's obtained in step 7, an average or expected curve can be obtained and error bands can be calculated which describe the uncertainty which arises from ignorance.

LOSS OF COOLING IN THE SPENT FUEL STORAGE POOL

The most serious failure in the spent fuel pit system would be the complete loss of water. This type of failure would result in the loss of cooling and removal of radiation shielding of the spent fuel and could result in the subsequent release of a significant amount of gaseous airborne radioactivity to the environment. The water in the spent fuel pit could be lost by the water being pumped, siphoned, leaked, or vaporized out of the pit.

The probability of a complete loss of water from the spent fuel pit by failures involving the pit cooling system is estimated to be orders of magnitude below accidents initiated by external events such as earthquakes. Ample sources of make-up water are available to mitigate this type of accident. The more likely accident sequences that could result in release of radioactivity to the environment involve loss of spent fuel pool cooling, heat-up and boiling of the spent fuel pit water, and a resulting release of radioactivity.

Prior to the transfer of spent fuel from the reactor to the spent fuel storage pool, water will enter the cladding of those pins which have experienced failure in operation. If, during the heatup of the pool, boiling occurs in the gap of these failed pins, it is likely that a release of radioiodine will occur to the pool water analogous to the spike that occurs at the time of reactor startup after a short shutdown period. Assuming that startup spikes are half as large as decompression spikes and that the most recently unloaded 1/3 of the core is affected, the release of iodine to the environment is represented by:

$$Q = \frac{W}{6M} \cdot t \cdot S \cdot f \cdot e^{-\lambda t_s} \cdot DF_w \cdot DF_f \qquad (1)$$

where W = steam generation rate from boiling pool
S = spike release of I-131 per percent clad defected in operating reactor
M = mass of water in pool
λ = decay constant
t_s = time since last shutdown
t = boiling time period
f = percentage of clad defects
DF_w = decontamination factor for pool
DF_f = decontamination factor for filter.

The analysis methodology was designed to account for the large number of dependencies among the factors affecting the estimate of probability versus release magnitude. Most of these factors are, however, dependent upon the time after refueling at which the accident (loss of pit cooling) occurs. For instance, the time after refueling (t_s) at which the accident occurs affects:

- The radioactive inventory of the fuel rods,
- The heat load to the pool, which in turn affects the time required to raise the fuel pit to boiling, and the boil-off rate,
- By virtue of the time to reach boiling, the probability of the pit reaching boiling,
- By virtue of the boil-off rate, the rate at which radioactivity is released to the atmosphere.

Therefore, the analysis methodology was structured around the dependencies inherent in the time after refueling at which the accident occurs. If it is assumed that the accident occurs on a given day after refueling, the relationships between the various factors affecting probability versus release magnitude can be defined and incorporated into the methodology. This suggests a partition of the sample space by day after last refueling, which was the approach taken. The probabilities versus release magnitude are estimated given that the accident occurs on each of the specified days after refueling, and the results are summed using the law of the total probability to obtain the probability versus release magnitude for the accident sequence.

A_i = the event that cooling is lost on the i^{th} day after refueling, and the accident results in pit boiling.

R_j = a specified release magnitude of size R_j.

By the law of total probability, the probability of the specified release magnitude greater than R_j, from this accident sequence is:

$$P\left[R_j\right] = \sum_{i=k}^{n} P\left[R_j/A_i\right] \; P\left[A_i\right] \qquad (2)$$

where $\sum_{i=k}^{n}$ refers to the summation over the n days between refuelings, starting with the k^{th}day (k=5 for the analysis reported herein).

For the i^{th} day after refueling, the model estimates the conditional probability of release greater than R_j given that loss of cooling occurs on the i^{th} day ($P[R_j/A_i]$), and the marginal probability that loss of cooling occurs on the i^{th} day ($P[A_i]$). These two probabilities are estimated by different techniques. The conditional probability is estimated by a partial failure analysis from a consideration of the factors that define the state of the system at the time of the accident, e.g., failed fuel percent, time required to repair the cooling system or mitigate the accident, etc. The marginal probability was estimated from a fault tree analysis of the fuel pit cooling system.

Both the marginal and conditional probability estimates are uncertain by virtue of the systematic error in the parameters upon which these estimates depend. The marginal probability estimate, $P[A_i]$, is uncertain due to the uncertainty in the failure rates of component failures leading to the event "the fuel pit begins to boil". The conditional probability, $P[R_j/A_i]$, is uncertain due to uncertainty in several physics and engineering parameters, namely, the iodine partition factor, and filter efficiencies in the hot, humid conditions that would result from a boiling spent fuel pit. Thus, both the marginal and conditional probabilities are more correctly thought of as distributions of possible probabilities for the accident sequence. These distributions reflect the best estimates of the systematic error in the parameters that are used to estimate the probabilities.

The probability of a specified release magnitude greater than R_j, $P[R_j]$, is thus also a distribution, since $P[R_j]$ is the sum of products of the conditional and marginal probabilities, both of which are distributions. To estimate the distributuion of $P[R_j]$, the model propagates the distributions of $P[R_j/A_i]$, and $P[A_i]$ to $P[R_j]$, using Equation 2. The model performs this error propagation for several specified values of R_j. The medians of the distributions for $P[R_j]$ define the centrality and shape of the probability versus release magnitude curve for the boiling fuel pit accident. The five and ninety-five percent confidence levels of the distributions for $P[R_j]$ serve to bound the probability versus release magnitude. The end product of the modeling effort is a complementary cumulative curve for probability versus release magnitude, and confidence bounds on the curve.

In Table 1, the sources of variation and their treatment in the partial failure analysis are described. In Table 2, the sources of uncertainty in the release magnitudes that result from ignorance are described.

RESULTS

The model was evaluated for an accident in the spent fuel storage pit of the PWR analyzed in WASH-1400. The pit was filled with three full cores, the results of nine refuelings conducted a half year apart. The risk per reactor year was evaluated for the period after the ninth refueling. Fig. 1 shows the median complementary cumulative distribution function for the release of I-131 to the environment. For comparison, the results for Class 9 accidents in WASH-1400 and an evaluation of Licensee Event Reports are also shown. The measure of consequences, curies of "equivalent I-131", involves a summation of radionuclides weighted by their health effects relative to I-131. Despite the large error bounds on the estimated CCDF, the results of the partial failure analysis indicate that this accident is not a major contributor to public risk for the plant analyzed.

For the example accident sequence examined in this study, the method of partial failure analysis was found to be a useful means of considering a spectrum of possible operational states in performing risk analyses. The differentiation which is made in the methodology between two sources of uncertainty, random and systematic, is felt to be particularly important. Not only does the mothodology produce a risk curve which includes the operational variation of parameters, but also provides confidence bounds on the curve.

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TABLE I. SOURCES OF VARIATION THAT INFLUENCE RELEASE MAGNITUDE

Source of Variation	Affects	Statistical Characterization
1) Time after refueling	Pool heatup rate; iodine spike	Uniform distribution between refuelings
2) Previously stored core sections	Pool heatup rate	Deterministic
3) Initial pool temperature	Time to boil	Normal distribution between 70 and 120°F
4) Failed fuel percent	Iodine spike	Lognormal distribution between 0.12 percent and 1 percent
5) Iodine spike given failed fuel percent	Iodine in water after spike	Empirical distribution from reactor spikes modified for spent fuel pool accident[3]
6) Cooling system repair time	Amount iodine released; probability of a boiling pit	Empirical distribution from WASH-1400, Appendix III

TABLE II. SOURCES OF IGNORANCE THAT INFLUENCE ESTIMATION OF PROBABILITY VERSUS RELEASE MAGNITUDE

Source of Variation	Affects	Statistical Characterization
1) Water DF	Iodine release rate	Lognormal distribution between .0016 and .16
2) Filter DF	Iodine released to atmosphere	Lognormal distribution between 99 and 99.99 percent

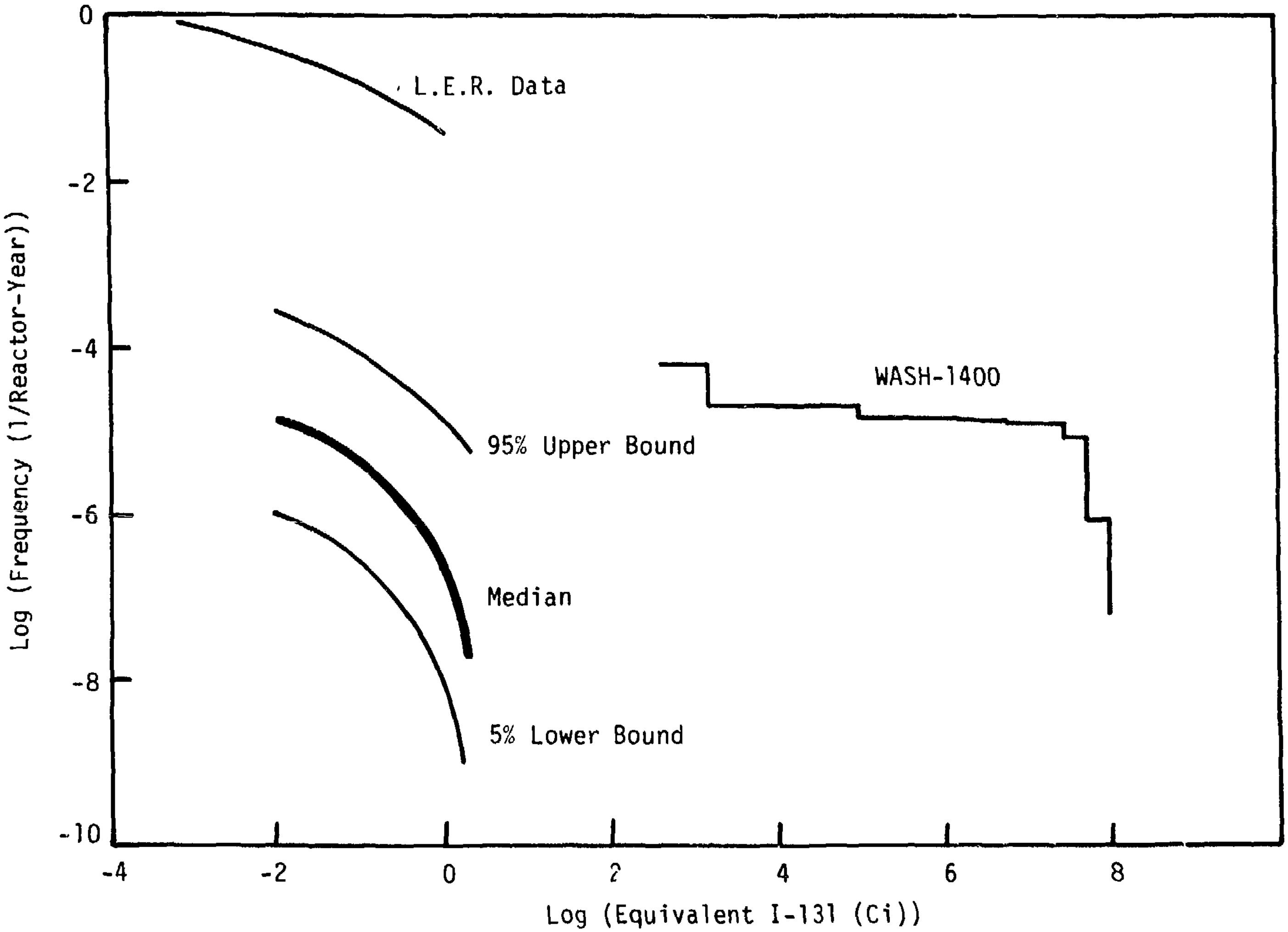

Fig. 1. CCDF for Iodine Release

Dup

THE LOFT AUGMENTED OPERATOR CAPABILITY PROGRAM

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ABSTRACT

The outline of the LOFT Augmented Operator Capability Program is presented. This program utilizes the LOFT (Loss-of-Fluid Test) reactor facility which is located at the Idaho National Engineering Laboratory and the LOFT operational transient experiment series as a test bed for methods of enhancing the reactor operator's capability for safer operation. The design of an Operational Diagnostics and Display System is presented which was backfit to the existing data acquisition computers. Basic color-graphic displays of the process schematic and trend type are presented. In addition, displays were developed and are presented which represent "safety state vector" information. A task analysis method was applied to LOFT reactor operating procedures to test its usefulness in defining the operator's information needs and workload.

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INTRODUCTION

A near consensus has been reached on the need to apply state-of-the-art technology to the safe operation problems of a commercial light water reactor (LWR) under upset or faulted conditions. The two major elements of this technology are: (1) computer technology and (2) functional analysis of operations.

COMPUTER TECHNOLOGY

Under off-normal operational conditions, the operator in a nuclear power plant is presented with an enormous amount of information which must be collected, processed, and evaluated in order to make appropriate control decisions as to whether the plant can be restored to normal operating conditions or should be shutdown.

Under emergency conditions, the active area of the control panel and the volume of raw data can exceed the saturation point of the operator. This data is presented to the reactor operator without prioritization in a short period of time. Yet, the operator needs more, not less, information concerning the status of crucial plant systems. Thereby, a dilemma exists in balancing a recognized need to reduce operator data overload against a perceived need by the operator for more data. This dilemma can be resolved by the use of computers to reduce raw information to significant information which can be displayed in recognizable form.

An Operational Diagnostics and Display System (ODDS) has been designed for use with the Loss-of-Fluid Test (LOFT) reactor at the Idaho National Engineering Laboratory. The ODDS is presently being evaluated during small break (loss-of-flow) tests conducted on the LOFT reactor. The ODDS will improve the operator's capability for making correct and timely control decisions.

LOFT is a scaled-down version of a commercial pressurized water reactor (PWR) (one sixty-fourth size). It is felt LOFT resembles a commercial PWR in man-machine factors which permit evaluation of computer-based graphic displays for their potential use in commercial LWR applications. The LOFT man-machine factors representative of typical LWRs are shown in Table I.

TABLE I
LOFT MAN-MACHINE FACTORS REPRESENTATIVE OF TYPICAL LWRs

1. Reactor Facility
    a. Nuclear Steam Supply System
    b. Main Control Room
    c. Automatic Protective Systems (RSS, ECCS, CIS)
    d. Instrument and Control Equipment
2. Operational Framework
    a. Technical Specifications
    b. Operating Procedures
    c. Operating Crew
    d. Training
    e. Maintenance Practices

DESIGN CONFIGURATION

The hardware components of the LOFT Operational Diagnostics and Display System (ODDS) are shown in Figure 1. The ODDS consists of a central processing unit (CPU), asynchronous multi-line controller (AMLC), memory unit, disk storage unit, magnetic tape unit, and display terminals. The CPU is a PRIME 550, a machine near the upper end of the performance range of minicomputers. The system is configured with 512 kilobytes of main memory and possesses two kilobytes of high speed cache memory to speed program execution. Both on-line and off-line storage capability are provided for the data files and programs. Three cathode ray tube (CRT) terminals provide an interface with the various users and user interaction with the system. The CRTs are RAMTEK devices interfaced with the PRIME by serial lines and are capable of graphics in eight colors. The same type of serial interface used with the CRTs is also used to connect the PRIME 550 with the LOFT Plant Log and Surveillance Subsystem (PLSS) computer through which data are dynamically acquired.

Initially, The ODDS has been configured to take advantage of the existing LOFT PLSS, a system built around a MODCOMP-IV computer already used to acquire plant information from process instruments in order to provide historical plant log and real-time monitoring functions. The software design approach with respect to data acquisition was to view the data as being comprised of two types: analog and event.

Analog data acquired by the PLSS are routinely buffered so a data point representing an average of several seconds of data for each analog channel is available for processing or presentation. Data transmitted from the PLSS to the PRIME are updated every five seconds. All analog data have been converted to floating point, engineering unit values before being sent to the ODDS.

Event data are discrete data which relate to a physical condition such as a breaker switch or valve position. They are updated to the ODDS every two seconds.

In keeping with the design approach of separating the event and analog data, each type of information is passed over a different physical line by an independent PLSS-resident program and is acquired by an independent program on the ODDS. Complexity of the communication process is kept to a minimum by use of a serial interface with all data transmitted at 9600 baud (bits per second).

Programs resident on the ODDS acquire data from the communication lines, reformat the data, and place the data into storage files on a disk storage unit. Analog and event data are each stored into circular files of approximately 10 hours duration. These data files may be spooled to tape for off-line storage and subsequent retrieval for replay purposes.

A package of display-oriented software exists which accesses the circular disk files and creates the various color displays seen by the user on the CRTs. At the heart of the display package is a set of routines known as the graphics display library. The application programs constructing the various displays all use the graphics display library.

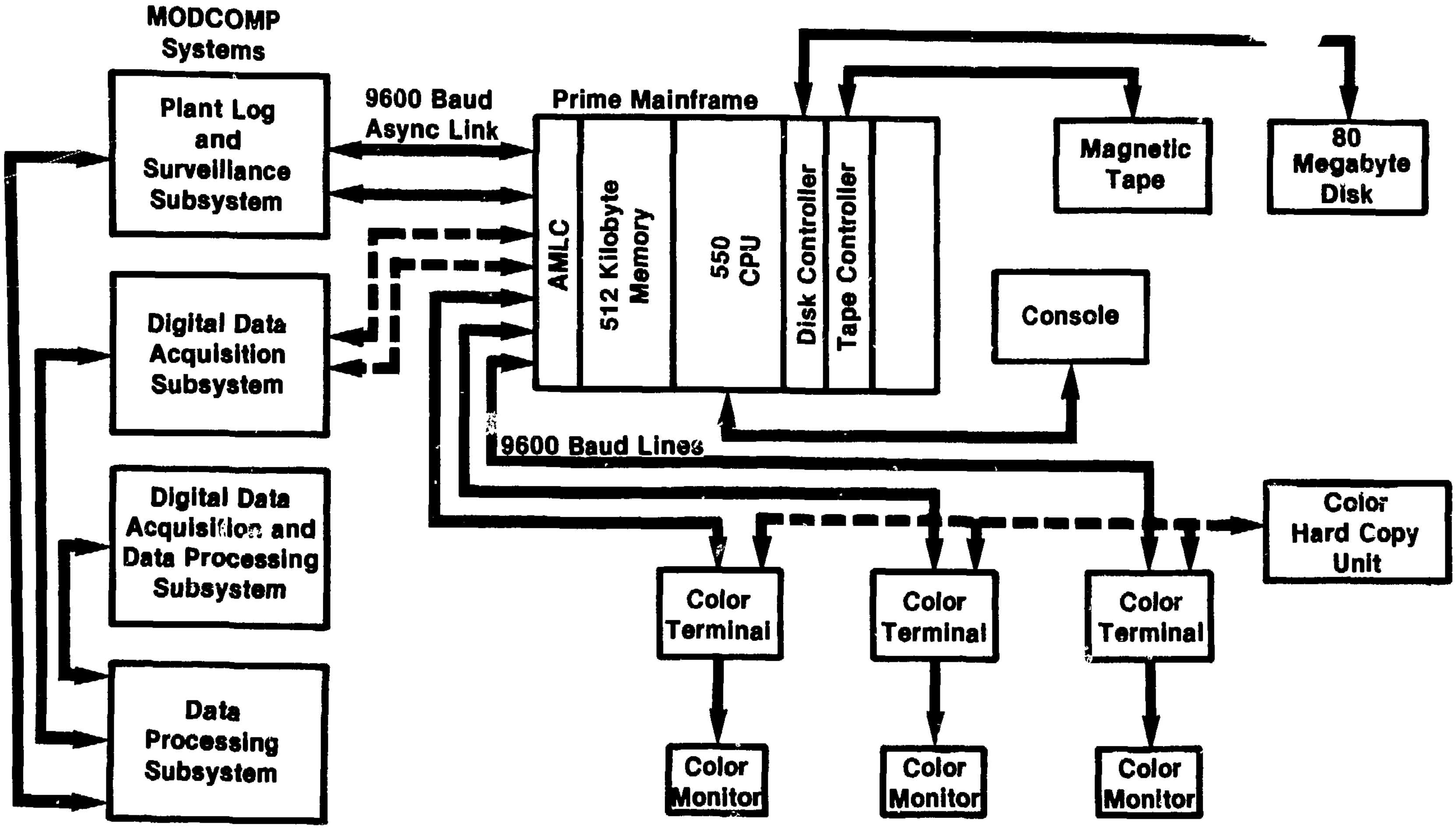

Figure 1. Operational diagnostics and display system (ODDS).

Expansion and enhancement of the software capability is planned. Some items under consideration are: (1) increased data update rates, (2) increased data base to support additional instrumentation, and (3) numerous new applications in the display program package.

BASIC DISPLAYS AND TREND INFORMATION AVAILABLE

A demonstration set of color graphic displays has been implemented on the LOFT ODDS. These displays were chosen to encourage immediate use of the ODDS by the reactor operator. Status-type displays were implemented first to get the ODDS into service rapidly (diagnostic or other complex programs take longer to design and implement). The general criteria used for the selection of LOFT displays were:

a. Displays should present information which is frequently used by the reactor operator during normal reactor operation,

b. Displays should also be of potential use in following the course of a small LOCA (loss-of-coolant accident) or operational transient,

c. Status-type displays should be implemented first,

d. Information should be presented in an integrated fashion to support specific plant evolutions or operation of crucial plant systems,

e. Displays should present information in formats which are complementary to those presently available for the conventional process instrumentation in use at LOFT, and

f. Baseline displays should use information derived from process (non-experimental) measurements.

The demonstration displays can be grouped into two sets: process schematics and status or trend plots. Process schematics exist for the primary coolant system, secondary coolant system and emergency core coolant system. These displays are simplified schematic diagrams with parameter values and component status (e.g., valve position) shown at the appropriate locations on the diagrams. Initial conventions are established for the representation of component status through the use of colors (e.g., pump on or off, vessel level) and symbol shape (e.g., valve open or closed).

Status and trend plots generally show three types of information: (1) present status of one or more crucial plant parameters, (2) recent past history of these parameters, and (3) operating limits for these parameters appropriate for the mode of operation for which the display was intended. Demonstration displays of this type include:

a. Plant heatup (actual vs technical specification limits)

b. Plant cooldown (actual vs technical specification limits)

c. Pressure vs temperature (hot leg conditions vs power operation limits)

d. Minimum pressure vs temperature (cold leg conditions, including pump operation limits)

e. General X-Y plot (any two parameters).

Typical demonstration displays of process schematic, safety state vector, and trend information available on the LOFT ODDS are shown as Figures 2, 3, 4, and 5. Small-break LOCA data from Experiment L3-2 are displayed.

Each of the baseline displays exists in two versions: a "control room operator" version and an "engineering" version. Each version of each display can be called up for viewing on any display terminal either by typing a simple mnemonic (e.g., "PCS" for the Primary Coolant System process schematic) or by pressing a special function key on the terminal keyboard. The control room operator displays have fixed formats and parameter ranges, and display only current data. The engineering displays allow the user to alter such features as the scaling of plots or the indicated status of components; they also allow the replay or display of historical information stored in the computer. This information base includes several hours of the most recent plant data as well as data from previous LOFT tests.

A number of limitations of the present display capabilities are recognized at this time. Some of the more significant ones are:

a. Development of display hierachy and structure has just begun; consequently the present displays are related only through the training and experience of the plant operator.

b. Nuclear industry standards for the use of color, symbology, and other display conventions for such systems have not been established.

c. Some information desired for the demonstration displays is not part of the available data base. (Over 60 status and parameter values have already been added to the LOFT data acquisition system to support the baseline displays.)

d. The displays can be regenerated at will by replaying historical data; however, no simulation capability presently exists to allow varying indicated plant status from that which actually occurred during LOFT operation.

FUNCTIONAL ANALYSIS OF LWR OPERATIONS

Task analysis is being used to determine the operator's information needs during normal and emergency operation of the LOFT facility. Task analysis is a systematic method for analyzing the operation of a system by (1) breaking the operation into its component parts and (2) extracting useful information concerning the operation of the facility. Task analysis is performed in four steps. First, the overall characteristics of system operation are examined to define relevant operating modes of the system and potential transfers between modes. Second, procedures are developed for

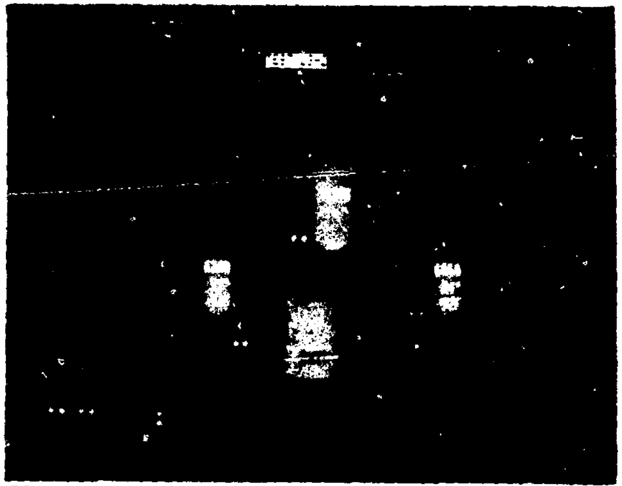

Fig. 2. Emergency Core Cooling System

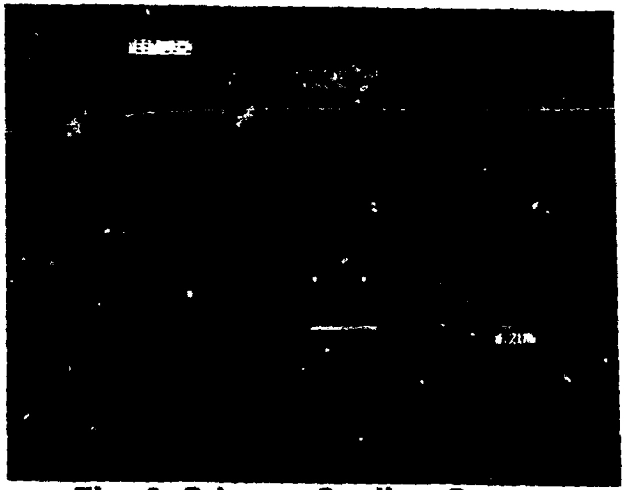

Fig. 3. Primary Cooling System

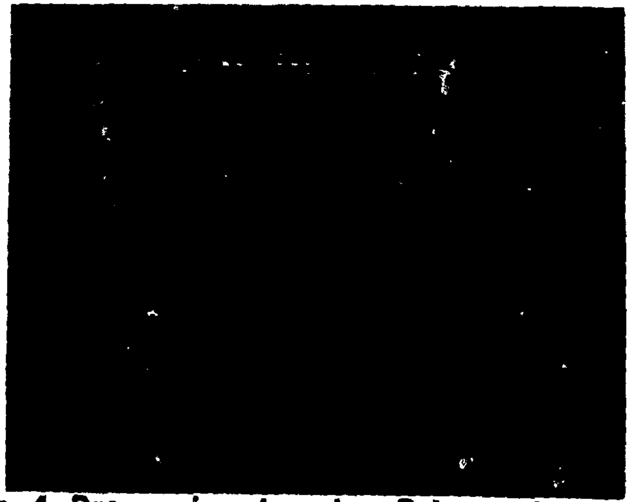

Fig. 4. Pressurizer Level vs Primary Pressure

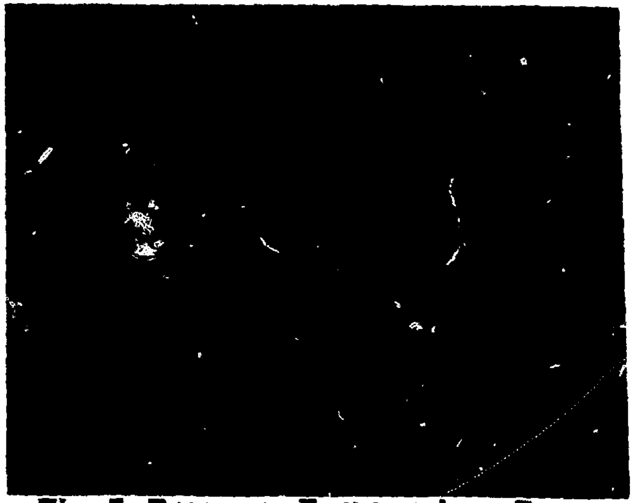

Fig. 5. Pressure-Temperature Curve

Note: The figures show actual data taken during LOFT small-break test L3-2. The figures are arranged left to right in time sequence.

each mode-to-mode transfer; the LOFT plant operating manual is being used as a basis for this step. Third, each procedure is flow charted to illustrate the operator's decision points and the potential paths through the procedure. Fourth, a tabular form is used to list information from the flow chart including: (1) required decisions, (2) information required to make the decision, (3) source of the information, (4) time available to act, (5) feedback associated with the correct action, and (6) alternative actions available if a malfunction occurs.

The results of LOFT task analyses are used: (1) to make recommendations to improve existing procedures and (2) to make recommendations for the design of CRT displays to be implemented on the ODDS. Representative results of this type of analysis are discussed in Reference 4.

CONCLUSION

The LOFT ODDS was placed in operation in January 1980 and was used by the reactor operators in conducting the LOFT L3-2 small-break test in February 1980. The ODDS is being readily accepted by the LOFT reactor operators as an aid in controlling the plant. Although only a limited number of baseline displays of process schematics and trend information are available at present, computer-based graphic displays are expected to gain acceptance in the future as a useful source of information to assist the reactor operator in his decision-making processes required for normal and off-normal reactor operations.

Functional analysis of operations appears to be as applicable to the LWR operational safety problems as to other modern man-machine control problems. Functional analysis and computer-based graphic technologies are being developed for the LOFT program to permit this unique facility to be used as a workshop and test bed for LWR operational safety problems.

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These references were used as definitions of where reactor operator capabilities should be augmented.

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PROBABILISTIC SAFETY ANALYSIS FOR SHIPPING LIQUEFIED PETROLEUM GAS IN THE VICINITY OF A NUCLEAR POWER PLANT

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ABSTRACT

A probabilistic safety analysis has been performed for truck and rail transportation of liquefied petroleum gas (LPG) in the vicinity of a nuclear power plant. Site-specific truck and rail accident data were combined with nationwide commodity-specific data to give a spill probability representative of local conditions. This information was used with cloud formation and atmospheric dispersion models to determine both the annual probability of excessive overpressure at plant structures and the annual probability of a flammable gas mixture at plant air intakes. Results are presented as a function of overpressure and plant-to-transportation route distance.

INTRODUCTION

For hazardous materials shipped along transportation routes near a nuclear power plant, it must be shown that the probability of adverse consequences resulting from accidental release of these materials is less than a specified maximum. One of the more frequently shipped hazardous materials classess is that of compressed LPG, propane, butane, etc. This paper focuses on LPG transported by truck and railroad tank bodies and evaluates the probability of the following events: (1) external overpressure exposure greater than a specified value on a safety-related structure resulting from explosions and (2) a flammable gas mixture occurring at the air intake to a safety-related structure. An analytical model is evaluated with typical input conditions and parameters so that the resulting family of curves represents a nominal power plant. These curves provide typical results that can be adjusted for conditions at a specific plant or used for siting studies before detailed site specific studies are performed

ANALYTIC MODEL

Plant exposure is initiated by transportation accidents involving LPG. An accident may or may not result in release of LPG. If LPG is released, it could lead to either an explosion or fire at the accident site, or to a drifting cloud. A drifting cloud is capable of an explosion or fire along its drift path if an ignition source is encountered and flammability conditions exist. If the drifting cloud reaches the plant before being ignited or before dropping below the lower flammability limit, it could be swept into an air intake where ignition could cause severe damage.

These accident consequences are used to define two regions surrounding the plant. Region I is defined as that area where, if a vapor cloud explosion occurs, the plant overpressure limit would be exceeded. If the overpressure limit is lowered, the region area expands. Also, the region is larger for railroad accidents because of the larger load capacity. Within Region I is a smaller area, Region II, which encompasses the safety-related air intakes. For this analysis, Regions I and II are assumed to be circular, with the plant at the center.

The probability of exposing the plant to an overpressure greater than a certain value is equal to the sum of the contributions from accident site explosions plus drifting cloud explosions. The former is the product of the number of shipments, and the probabilities of an accident per shipment mile, of a spill given an accident, of an explosion given a spill, and the length of the route in miles within Region I. For the latter, a drifting cloud explosion, the probability is the product of the number of shipments, and the probabilities of an accident per shipment mile, of a spill given an accident, of a drifting cloud forming if a spill occurs, of the wind blowing towards Region I, of ignition after the cloud reaches Region I, of an explosion given ignition, and the length of the route outside Region I that is within the radius to the lower flammable limit. The length of route is divided into small segments to account for the variation of distance and direction to the plant.

The range of an explosion that will result in a pressure greater than a specified value (that is, the size of Region I) is calculated from plant geometry and an explosion overpressure range scale law[1]. For an unconfined vapor cloud explosion, a TNT equivalent with 10 percent energy yield is used. The entire quantity in the cloud, which was taken to be the isenthalpic flash fraction, was assumed to be involved in the fuel air reaction. The change in the quantity of vapor between upper and lower flammable limits as the cloud disperses was conservatively neglected.

The probability of a flammable cloud reaching the plant is the product of the number of shipments and the probabilities of an accident per shipment mile, of a spill given an accident, of wind blowing to Region II, and of nonignition before reaching Region II. The flammable cloud size, as determined by the dispersion analysis, was considered in the analysis as effectively increasing the size of Region II.

The dispersion model used is an instantaneous puff model modified to account for initial gravity slumping of heavier-than-air vapor. Initial dispersion parameters are determined from the Van Ulden gravity spreading model.[2] The initial cloud formed at the accident site is assumed to be cylindrical, with the axis perpendicular to the ground, and it spreads according to the density difference between the cloud and the air. It is assumed that during the gravity spreading phase, the flammable vapor concentration in the cloud remains unchanged. The cloud spreads until the turbulent energy of the spreading equals the potential energy difference between the heavy gas layer and the surrounding air. At the end of the gravity spreading, the concentration of the cloud is assumed to have a Gaussian distribution with the center point concentration being pure vapor. Normal Gaussian dispersion then occurs, with cloud standard deviations being determined from the initial values at the end of gravity spreading and those values corresponding to the assumed Pasquill stability category. It should be noted that the dispersion model is used only to determine the maximum downwind distance to the lower flammable limit and to the lateral cross-wind size of the cloud.

As previously indicated, a flammable vapor cloud is swept into a plant air intake only if the gas cloud reaches the plant above the lower flammable limit concentration without prior ignition. The probability of prior ignition is based on the historical data of LPG spill accidents, and is fitted by a curve as a function of distance from the accident site.[3] The curve agrees reasonably well with statistics quoted by James[4] where for 81 vapor cloud ignitions, 58 percent occurred from a few feet up to 50 feet of the accident site, 18 percent between 50 and 100 feet, and 24 percent from 100 to 300 feet. The data in reference 3 also indicates that 10.5 percent of the drifting cloud ignitions resulted in an explosion, while 89.5 percent resulted in a fire.

ACCIDENT DATA

Accident data for the model evaluation were derived from applicable sources. Site specific accident rates were combined with nationwide experience to evaluate annual frequency by severity of LPG accidents for both highway and railroad.

Local truck accident rates were assessed from data collected over a 4-year period from a 10-mile section of interstate highway passing a particular power plant. An accident was counted if there was personal injury, death, or property damage in excess of $2,000. Trucks were defined as those greater than 5,000 pounds, but not pickup trucks, vans and buses. During the period of the study, there were 84×10^6 truck miles observed and 20 accidents reported yielding an accident rate of 0.24×10^{-6} accidents per truck mile. Nationwide experience indicates that the tank truck accident rate is approximately 55 percent of the accident rate for all trucks. Hence, the local tank-truck accident rate was taken as 0.13×10^{-6} accidents per mile.

The spill probability given an accident was assessed from nationwide accident reports submitted to the Bureau of Motor Carrier Safety of the Department of Transportation (DOT). Truck accidents for a five 5-year period were examined. Of the approximately 30,000 accidents reported each year, those that involved tank truck bodies, intercity trips, divided highways (excluding interchanges), and transportion of pressurized liquefied flammable gases were selected. These characteristics were chosen to represent LPG transport on U.S. interstate highways. Overpasses were excluded because there are none near the specific plant being considered. During the 5-year period, there were 109 accidents that had these characteristics 7 of which involved spills. This yields 6.4 percent chance of a spill given an accident that satisfies the above conditions.

The severity of an accident with a spill of a hazardous material was assessed from data available at the Office of Hazardous Materials of the DOT. This data was analyzed to develop a distribution of spill quantity; for example, 50 percent of the spills were less than 5,000 gallons and 30 percent less than 250 gallons.

If a tank-truck accident with a spill has occurred, three things can happen: (1) the spilled LPG can vaporize and expand, forming a drifting cloud, or (2) the spilled LPG can be ignited forming a fire (the size of the fire can vary from small to large), or (3) the spilled LPG can expand and be ignited, causing an explosion. LPG spill data have been examined to determine and classify the results of the spill into one of the three classifications above. Of the 23 LPG spills occurring in the 5-year period ending in 1977, 12 were not ignited and there were no explosions leading to significant overpressure. Preliminary incomplete information on one accident that happened in 1978 in Mexico indicated an explosion with potentially significant overpressure occurred. This combined data indicates that 50 percent of spills will not be ignited and 4.2 percent of spills would lead to an explosion.

Local railroad accident rates were evaluated from data collected over an 11-year period for a 100-mile section of track typical of the one passing the power plant. During this period, there were 2.7×10^6 train miles and 10 accidents yielding a train accident rate of 3.7×10^{-6} accidents per train mile (a train consists of engines and associated cars). The nationwide average accident rate for a coincident 10-year period is 11×10^{-6} accidents per train mile. Additionally, nationwide data of pressurized liquefied flammable gases showed a rate of 0.15×10^{-6} loss-of-lading accidents per loaded tank car mile[5]. The nationwide loss-of-lading rate was adjusted to a local rate by the ratio of local to nationwide train accident rates, yielding 0.051 x 10^{-6} loss-of-lading accidents per loaded tank-car mile.

In a loss-of-lading accident, various outcomes are possible depending on the amount spilled, the presence of an ignition source, and the flammability conditions. Using data from the Office of Hazardous Materials, DOT, 163 accidents were classified and probabilities were evaluated. The results showed that 51 percent of the spills were not ignited and 1.8 percent of the spills resulted in a fuel air ignition explosion. Data on 76 LPG tanks car spills were analyzed to determine spill quantity distribution. This indicated that about 25 percent of the spills released less than 12,000 gallons and 20 percent of the spills released less than 1,000 gallons.

RESULTS

Results for typical rail and highway shipment conditions are presented in Figures 1 and 2. A summary of input information is provided in Table I. The results are annual probabilities of (1) exceeding a specified peak reflected overpressure or (2) flammable gas reaching the plant. In Figure 1, the probabilities are shown as a function of distance between the plant and the transportation route. It shows that the probabilities, especially for overpressure, decrease rapidly as the distance between the plant and the transportation route increases. When the distance exceeds the radius of Region I (that is, when an accident site explosion would not exceed the overpressure criteria), the probability of unacceptable overpressure drops below about one-fifth to one-tenth of the close-in value. Drifting cloud explosions are therefore relatively minor contributors to the risk.

For the same number of shipments, the probability from highway transportation is lower than that from rail by one to two orders of magnitude. This results from the smaller quantity shipped by highway and the somewhat lower highway spill rate. The effect of allowable overpressure is shown in Figures 1 and 2. At close-in distances, the annual probability shows little change with respect to overpressure changes, but as the distances increases, overpressue has a greater effect.

Results of sensitivity analyses are shown in Table II. As can be seen, the results are not particularly sensitive to the variations considered, except for the flammable cloud probability which that is sensitive to the drifting cloud ignition probability. The alternative model of an ignition probability of 10 percent for each 20 meters of cloud travel results, for example, in an ignition probability of 0.38 in 300 feet compared to the base case value of about 0.83 in the same 300 feet. The flammable cloud intake probability remains less limiting than the external overpressure probability.

CONCLUSIONS

From the results presented above, site-specific conclusions can be drawn as to the necessary separation distance between transportation routes and a nuclear power plant in order to keep the probability of adverse effects of shipping LPG below a desired value. The results are considered to conservatively represent the probability of unacceptable effects for either an interstate highway or mainline railroad without unusually hazardous characteristics.

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TABLE I

Summary of Input

Wind direction frequency	Uniform
Radius of the plant region (Region II)	200 ft
Number of Shipments	100 per year
Drifting cloud	
Fire/ignition	0.895
Explosion/ignition	0.105
TNT energy equivalent yield	10%
Atmospheric stability class	G
Wind speed	1.5 m/sec

	Mainline Railroad	Interstate Highway
Maximum shipment quantity (gallons)	30,000	10,000
Accident probability (Per mile)	$0.51x10^{-7}$	$0.13x10^{-6}$
Probability of spill given an accident	*	0.064
Probability of accident site fire given a spill	0.47	0.46
Probability of accident site explosion given a spill	0.018	0.042

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* Included in accident probability.

TABLE II

Results of Sensitivity Study

Parameter	Probability of Exceeding 3 psi (per year)	Probability of Flammable Vapor Cloud Being at the Plant (per year)
Base case*	0.14×10^{-6}	0.022×10^{-6}
Flash fraction - 0.5	0.17×10^{-6}	0.025×10^{-6}
Yield, 0.2	0.20×10^{-6}	0.023×10^{-6}
Drifting cloud explosion probability, 0.2	0.23×10^{-6}	0.022×10^{-6}
Drifting cloud ignition probability, 10% per 20 meters	0.13×10^{-6}	0.070×10^{-6}

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* The parameters for base case are flash fraction, 0.35; Yield, 0.1; Drifting cloud explosion probability, 0.1. For the drifting cloud ignition probability the curve is derived from Reference 3. Distance between the center of the plant to the railroad is 1000 ft.

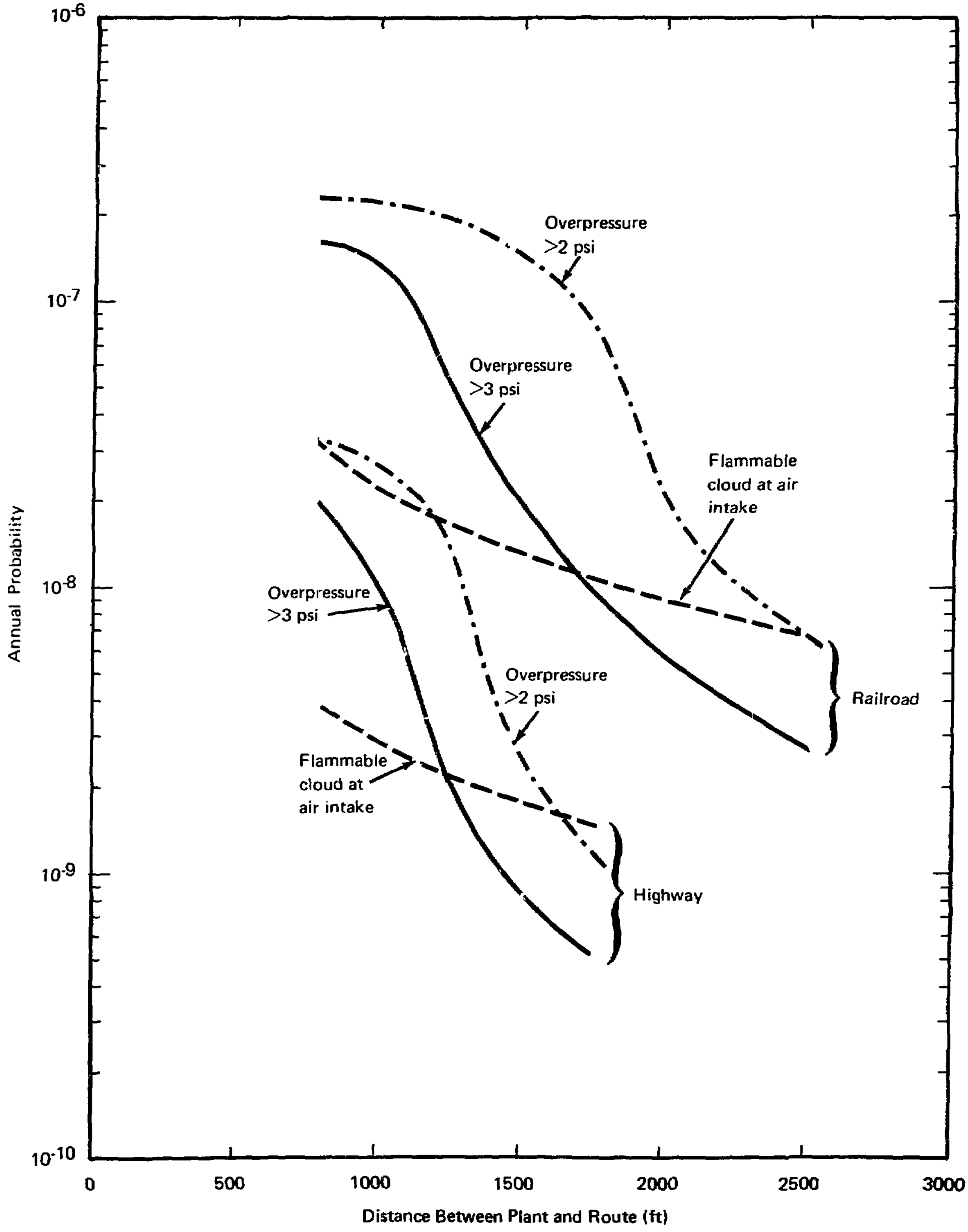

Figure 1 Probability of Exceeding Specified Overpressure and of Flammable Gas Cloud at Air Intakes (100 annual shipments of LPG. Allowable peak reflected overpressure of 3 psi or 2 psi)

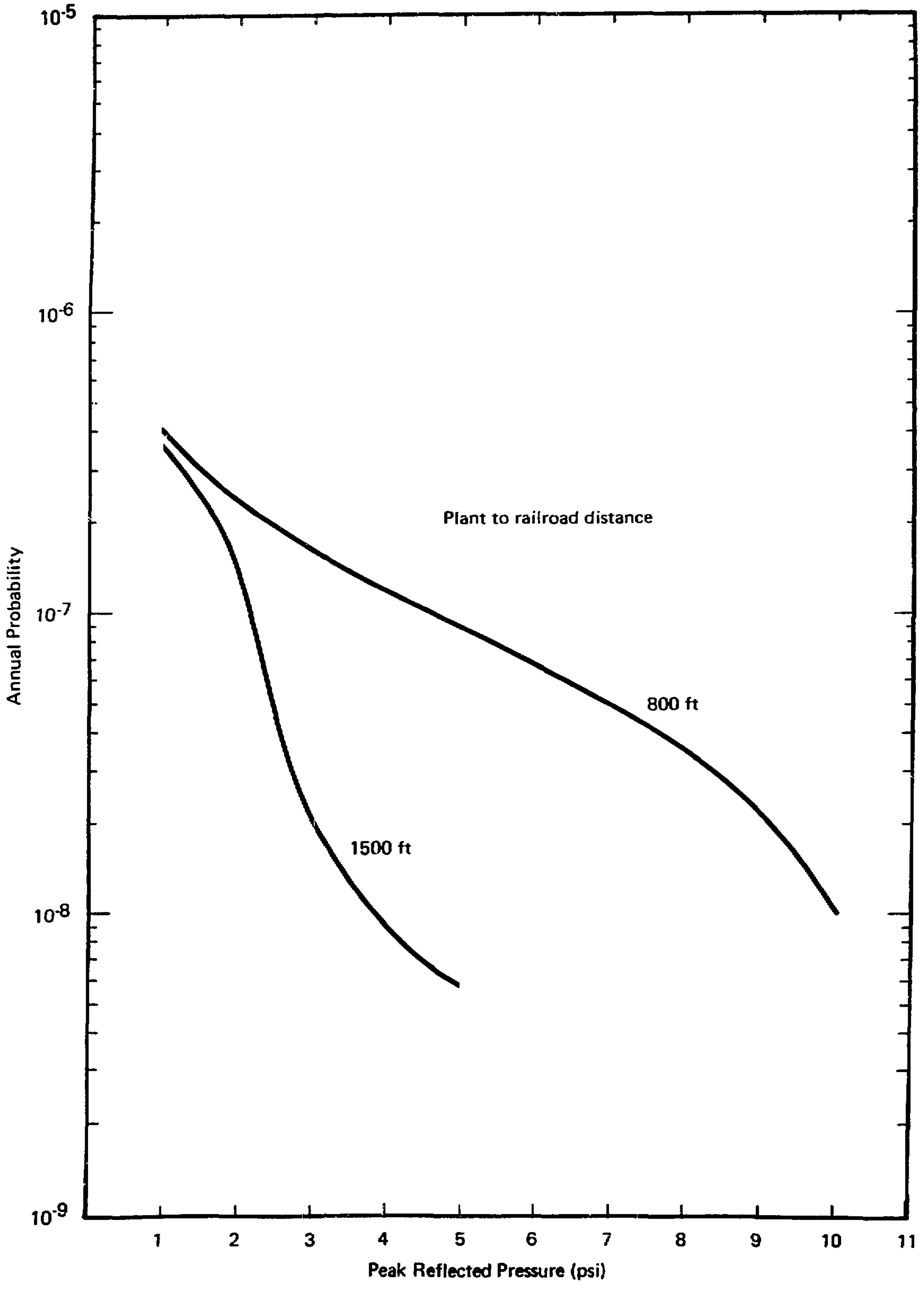

Figure 2 Probability Versus Overpressure Damage Criteria (100 Rail Shipments)

THE USE OF SAFETY FUNCTIONS
IN EMERGENCY PROCEDURE GUIDELINES

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ABSTRACT

A nuclear power plant can be thought of as a single system with two major subsystems: equipment and people. Both play important roles in nuclear safety. Whereas, in the past, the role of equipment had been emphasized in nuclear safety, the accident at Three Mile Island (TMI) and its subsequent investigations point out the vital role of the operator. This paper relates the operator's role of mitigating events to the concept of safety functions.

INTRODUCTION

Much investigation [1-4] and reflection has been done since TMI to improve equipment and operator performance to make such an event even less likely in the future. During the course of the Three Mile Island (TMI) event and subsequent investigations, frequent reference was made to the operator's "mindset" during the accident. [1] The inference was that the operator's training and experience had not prepared him to fully recognize the situation that was unfolding in from of him.[1,2] His "mindset" caused him to ignore or reject certain information that was essential for him to analyze the situation properly and take timely corrective action.

The designer's "mindset" of the operator's role in plant safety was also reviewed. Designers make assumptions of the operator's role, both during normal plant operations and during plant accidents. This information must be conveyed to the operator in a practical form. One method used to accomplish this has been through the plant operating and emergency procedures guidelines.

The intent of this paper is to relate the operator's role in mitigating events as conveyed in emergency procedure guidelines to the concept of "safety functions." Accomplishment of safety functions prevent core melt or minimize radiation releases to the general public. They can be used to provide a hierarchy of practical plant protection that can be transmitted from the designer to be used by an operator.

Assisting installed equipment in the accomplishment of safety functions is the operator's role in mitigating events. He needs to monitor the plant to verify that the safety functions are accomplished. In addition, he has to actuate those systems that are not fully automated and intervene where the automatically actuated systems are not operating as intended. There are three prerequisites to the fulfillment of this role:

1. information that identifies the plant state,
2. procedures that cover the situations encountered during events, and

3. comprehensive training to use the information and procedures to best advantage in responding to events.

Application of the concept of safety functions can be used to make significant improvements in each item above.

DISCUSSION

Safety Functions

The operator needs a systematic approach to mitigating the consequences of an event. The concept of "safety function" introduces that systematic approach and presents a hierarchy of protection. Safety functions are accomplished by performing mitigating actions. Actions may result from automatic or manual actuation of a system (reactor protection system generates a trip, operator aligns the shutdown cooling system), from passive system performance (safety injection tanks feed water to the reactor coolant system), or from natural feedback inherent in the plant design (control of reactivity by voiding in the reactor).

There are ten safety functions needed to mitigate events and contain stored radioactivity.

Safety Function	Purpose
Reactivity Control	Shut reactor down to reduce heat production
Reactor Coolant System Inventory Control	Maintain a coolant medium around core
Reactor Coolant System Pressure Control	Maintain the coolant in the proper state
Core Heat Removal	Transfer heat from core to a coolant
Reactor Coolant System Heat Removal	Transfer heat from the core coolant
Containment Isolation	Close openings in containment to prevent radiation releases
Containment Temperature and Pressure Control	Keep from damaging containment and equipment
Combustible Gas Control	Remove and redistribute hydrogen to prevent explosion inside containment
Maintenance of Vital Auxiliaries	Maintain operability of systems needed to support safety systems
Indirect Radioactivity Release Control	Contain miscellaneous stored radioactivity to protect public and avoid distracting operators from protection of larger sources

In all safety functions, the word control means accomplishment of the safety function such that core melt is prevented or radioactive releases are kept within acceptable limits. Control involves manual or automatic actuation of equipment, or the natural passive capabilities built into the plant.

There are five anti-core melt safety functions: Reactivity control, reactor coolant system (RCS) inventory control, RCS pressure control, core heat removal, and RCS heat removal. The purpose of the first anti-core melt safety function, reactivity control, is to shut the reactor down and keep it shut down, thereby reducing the amount of heat generated in the core. Reactivity is controlled in the short term by insertion of the control rods and/or through the natural feedback mechanisms of voiding in the reactor coolant. In the long term, reactivity is controlled by the addition of borated water to the reactor coolant system. Borated water can be added to the reactor coolant system using the charging and boric acid addition portions of the chemical and volume control system, the high and low pressure safety injection system and/or the safety injection tanks.

The purpose of the second and third anit-core melt safety functions, reactor coolant system (RCS) pressure and inventory control, is to keep the core covered with an effective coolant medium. RCS pressure control can involve either pressure maintenance or pressure limitation. Likewise, RCS inventory control can involve either inventory maintenance or inventory limitation. Under normal circumstances, RCS pressure and inventory control is maintained automatically by the pressurizer pressure and level control systems in conjunction with the reactor coolant system pressure boundary. These systems use the pressurizer spray valves and the letdown system to control pressure and inventory respectively, and they use the pressurizer heaters and charging system to maintain pressure and inventory respectively. If the pressure and level control systems are unable to limit RCS pressure and inventory, the pressure and inventory can be kept within bounds by action of the primary safety valves. In the event that RCS inventory and/or pressure becomes inappropriately low due to an opening in the reactor coolant pressure boundary or excessive cooling of the reactor coolant system from excess steam flow, RCS inventory is maintained by injection of borated water by the safety injection system or the safety injection tanks.

The purpose of fourth anit-core melt safety function, core heat removal, is to remove the heat generated in the core by radioactive dacay and transfer it to a point where it can be removed from the RCS to prevent the fuel from melting. This is accomplished by passing a coolant medium through the core to a heat removal point. Normally, the reactor coolant pumps are used to provide forced reactor coolant flow through the reactor core to the steam generators. In the absence of forced reactor coolant flow, the core can still be cooled by a natural circulation induced by a temperature differential from the steam generators to the core. (This implies that the steam generators must be available to act as a heat sink). If natural circulation cannot be established, heat can be removed from the core by boiling and movement of the steam to a point such that it can be discharged through an opening in the reactor coolant system piping.

The final anti-core melt safety functions is RCS heat removal. The purpose of this safety function is to transfer heat from the core coolant to another heat sink. If this is not done, core heat removal will not be possible. RCS heat removal is normally accomplished by transferring heat from the reactor coolant to the secondary system in the steam generator. The secondary system water is supplied by the main feedwater system or the auxiliary feedwater system. Reactor coolant heat can be transferred to the component cooling water via the shutdown cooling heat exchanger, provided that the reactor coolant system pressure is less than the shutdown cooling system pressure interlock setpoint.

If no other heat sink is available, reactor coolant system heat removal can also be accomplished by discharging the hot reactor coolant directly into the containment through a pressure boundary opening or a primary relief valve.

The foregoing discussion of the five anti-core melt safety functions illustrates that each safety function can be accomplished by a multiplicity of systems, and, in addition, many of the systems support more than one safety function. Under some circumstances, the execution of one safety function causes another safety function to be accomplished. Particular methods of accomplishing one safety function sometimes facilitate and sometimes prevent a particular method of accomplishing another safety function. This interaction, or synergy among the safety functions is an important feature of this concept.

Three safety functions contribute to containment integrity: containment isolation, containment pressure and temperature control and combustible gas control. The primary objective of these safety functions is to prevent major radioactive release by maintaining the integrity of the containment structure. Accomplishing the first safety function, containment isolation, assists in maintaining containment integrity by ensuring that all normal containment penetrations are closed off. Containment isolation is accomplished by sensors for measuring containment pressure, electronic equipment to generate and transmit an isolation signal when the containment pressure exceeds a setpoint, and a set of valves for isolating each containment penetration. (These valves are generally part of other systems also.) Each containment penetration is provided with two isolation valves, one inside containment and one outside containment.

The purpose of the second containment integrity safety function, containment temperature and pressure control is to prevent overstress of the containment structure and damage to other equipment from a hostile environment by keeping containment pressure and temperatures within prescribed limits. Containment pressure and temperature are controlled using the containment spray system and the containment cooling system.

Likewise, combustible gas control, the third containment integrity safety function, is needed to prevent containment overstress caused by explosion of hydrogen gas inside containment. The hydrogen would evolve from the metal-water reaction in the event of failure of one or more of the anti-core melt safety functions. Hydrogen gas is removed from the containment atmosphere by the hydrogen recombiners. The containment spray system and the fan coolers can also help in combustible gas control by redistributing the hydrogen gas throughout containment, thus preventing the formation of flammable pockets of hydrogen gas.

The purpose of the control of indirect radioactivity release safety function is to prevent radioactive releases from sources outside containment. These sources include the spent fuel pool and the radioactive waste storage facilities (gaseous, solid and liquid, including radioactive coolant). The systems used to control releases from these sources include the radiation monitoring system, the spent fuel pool cooling system and the waste management and processing systems.

The last safety function is maintenance of vital auxiliaries. The systems used to accomplish the nine safety functions discussed above are all supported by various auxiliary systems. These auxiliary systems provide such services as instrument air needed for opening and closing valves, electric power for running pump motors and operating instruments, and an ultimate heat sink to which RCS and core heat can be transferred. Vital auxiliaries must be maintained in order to successfully accomplish the other safety functions.

Each anti-core melt safety function has priority relative to the others as shown in the figure. In general, reactivity control is the foremost function because the amount of heat that must be removed from the core is determined by how well this function is accomplished. Next in precedence are those functions for appropriately maintaining a core cooling medium. To achieve this, actions must be accomplished to maintain an adequate reactor coolant system inventory and an appropriate reactor coolant system pressure. Finally, if core heat removal is not carried out, the reactor coolant system heat removal is irrelevant. Not only should the operator keep this hierarchy in mind, but he should also recognize the need for the vital auxiliaries to carry out these safety functions.

Multiple Success Paths

Nuclear power plants are designed so that there are two or more ways that can be potentially used to accomplish safety functions. That is, for each safety function there are several possible success paths. In general, the effectiveness of a particular success path for accomplishing a safety function depends upon what systems are operable in the plant and on whether or not the process variables are within the design range of the particular system or subsystem that will be used. In other words, the method of accomplishing a safety function depends on the plant state at the time the function is to be executed. The state that exists at the time of an event is affected by the event and by manually and automatically actuated system actions.

To accomplish the safety functions, the operator does not need to know what event has occurred. He does, however, need to know what safety functions must be accomplished, what success paths are possible and the conditions of the plant. This information defines the state of the process variables and the state of the plant equipment. The plant state can be correlated to the appropriateness and availability of the various success paths for a given safety function. The means of mitigating an event depend on the plant state produced by the event.

Combustion Engineering currently uses Sequence of Events Diagrams (a derivative of Safety Sequence Daigrams [5]) to represent the success paths available to mitigate an event. Sequence of Event Diagrams are intended to illustrate the possible ways to accomplish each safety function challenged during a particular event. These diagrams present the success paths with the appropriate system actions. Both automatic and manually actuated system actions are shown. Sequence of Events Analysis is currently part of the NRC specifications for Safety Analysis Reports, [6] and therefore are presented in the FSARs for St. Lucie Unit 2 [7] and for our System 80 [8] Standard Plant. Combustion Engineering has also used Sequence of Events Analyses as a design review tool for San Onofre Units 2 & 3 [9], Forked River, and St. Lucie Unit 2. In performing this work, it has been found that this type of technique is essential for the understanding of how the operator and plant systems work together to mitigate the consequences of events and are a vital primary input to emergency procedure guidelines.

Use of the Safety Function Concept to Assist the Operator

The safety function concept, which incorporates the principles of safety function hierarchy and multiple success paths dependent on the plant state, can help the operator fulfill his role of assisting the plant systems to mitigate the consequences of an event. In order to assist in accomplishing the safety functions he needs the following:

1. Sufficient and intelligible information about the plant state

2. Comprehensive procedures prescribing preferred and alternate success paths for each safety function

3. Adequate training in the concept and execution of safety functions

(Note: The operator does not need to know the initiating event as long as he can determine the plant state and therefore determine the safety functions in jeopardy.)

Currently, an operator responds to an event by following one of the event specific emergency procedures (based on emergency procedure guidelines) of which there are N. If none of the event specific procedures apply, he resorts to an unwritten (N+1)st (pronounced "N plus first") procedure. Simply stated, he does what he thinks he should to mitigate the event. The safety function concept and making the procedures plant state dependent can be used to improve the N current emergency procedures. In addition, these approaches can be used to develop a documented version of the (N+1)st procedure.

Using the safety function concept, the individual emergency procedure guidelines can be standardized. A typical event specific procedure guideline would identify, for a given set of plant symptoms, what safety functions must be accomplished, what automatic systems are available to accomplish them, which backup systems must be actuated if the automatic systems fail, and what the expected plant response is. The safety function concept can also be used to handle, in a single procedure guideline, events which produce similar plant states.

The operator's actions during an event depend on the safety functions which need to be accomplished and the success paths which can be used. The operator determines this by the symptoms (either for an event or for a range of plant states). For the situation where either none of the previously developed procedures apply or the plant did not respond as expected, the (N+1)st procedure whould provide the operator with both a set of guidelines to identify the safety functions in jeopardy and the success paths available, and a checklist for assuring that all safety functions are accomplished. All procedures should reflect the safety function concept. The main benefit of this approach is that guidance will be provided for all eventualities.

In structuring operator training, the safety function concept is meaningful because it contributes to a more comprehensive awareness of how the plant functions as a unit and how the various systems work together to accomplish each safety function. Not only will this awareness help the operator mitigate the consequences of an event, but it will help him set up and operate the plant in such a manner that the frequency and severity of the initiating events will be reduced. (See discussion of these other operator roles in Reference [10]).

SUMMARY

One of the operator's roles in nuclear plant safety is to assist in mitigating the consequences of adverse events. In mitigating an event, the operator can use emergency procedures structured around the concept of safety functions.

A safety function is defined as something that must be accomplished to prevent core melt or to minimize radiation releases. There are ten safety functions. Safety functions should be viewed as having a certain priority, i.e., some safety functions should have precedence over others in the operator's mind.

This does not mean that any safety functions are unimportant. All safety functions are important. Nor does it mean that maintenance of one safety function will inherently carry out an arbitrary other safety function. All safety functions must be carried out. There are several possible success paths for accomplishing each safety function. The availability or appropriateness of a given success path for mitigating an event depends on the existing plant conditions.

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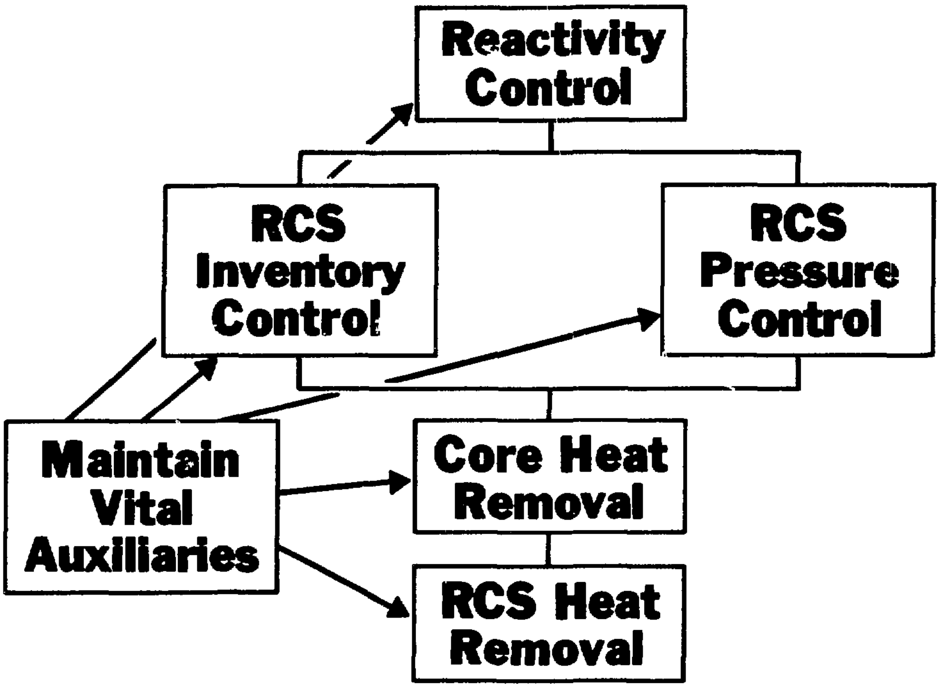
ANTI-CORE MELT SAFETY FUNCTION HIERARCHY



SESSION XXII

FISSION PRODUCT TRANSPORT

Chairmen

D. A. Nitti - Babcock and Wilcox

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Dup

FISSION GAS RELEASE IN LWR FUEL MEASURED DURING NUCLEAR OPERATION[a]

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ABSTRACT

A series of fuel behavior experiments are being conducted in the Heavy Boiling Water Reactor In Halden, Norway, to measure the release of Xe, Kr, and I fission products from typical light water reactor design fuel pellets. Helium gas is used to sweep the Xe and Kr fission gases out of two of the Instrumented Fuel Assembly 430 fuel rods and to a gamma spectrometer. The measurements of Xe and Kr are made during nuclear operation at steady state power, and for ^{135}I following reactor scram.

The first experiments were conducted at a burnup of 3000 MWd/t UO_2, at bulk average fuel temperatures of ~850 K and ~23 kW/m rod power. The measured release-to-birth ratios (R/B) of Xe and Kr are of the same magnitude as those observed in small UO_2 specimen experiments, when normalized to the estimated fuel surface-to-volume ratio. Preliminary analysis indicates that the release-to-birth ratios can be calculated, using diffusion coefficients determined from small specimen data, to within a factor of ~2 for the IFA-430 fuel. The release rate of ^{135}I is shown to be approximately equal to that of ^{135}Xe.

INTRODUCTION

Measurement of the release of fission product gases and volatiles from UO_2 during nuclear operation provides data which can be used in the development and assessment of models for predicting fission gas release, in the assessment of possible contributions of fission product volatiles (iodine) to stress corrosion cracking induced fuel rod failure, and in establishing the inventory available for release in the event of a breach in the fuel cladding.

Detailed and extensive studies of the release of fission gases and volatiles from small specimens of UO_2 have been performed both in the U. S.[1,2] and in the U. K.[3,4] From these studies the release to birth ratios (R/B) and diffusion coeffecients for Xe, Kr, and I have been determined and, as a consequence, the release of fission gases and iodine from small samples of UO_2 is predictable.

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[a] Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.

[b] OECD Halden Reactor Engineer

The release of fission product gases and volatiles from full size UO_2 fuel pellets in typical LWR design fuel rods is being investigated by EG&G-Idaho using the Instrumented Fuel Assembly 430 (IFA-430) in the Halden Reactor located in Halden, Norway. This work is being performed as part of the U.S. Nuclear Regulatory Commission's Reactor Safety Research Program. This paper presents the preliminary results and analysis of experiments performed with IFA-430 to characterize the release of Xe, Kr, and I fission products during steady state nuclear operation at peak fuel centerline temperatures below 1250 K. The release to birth ratios (R/B) are presented, the possible release mechanisms discussed and the results are compared with data from small sample UO_2 experiments.

EXPERIMENT DESIGN AND CONDUCT

IFA-430 contains four fuel rods (1.28-m-long) with 10% enriched UO_2 fuel pellets of 95% theoretical density. Two rods, termed gas flow rods, each have a centerline thermocouple and three axially distributed pressure transducers mounted directly to the cladding to measure internal gas pressure. These two fuel rods are connected to a gas flow system, shown in Figure 1, which allows the released fission gases to be swept out of the fuel rods to a gamma spectrometer where they are quantitatively measured. The other two rods are equipped with two centerline and three off-center thermocouples and are sealed and pressurized with helium to 0.48 MPa.

The IFA-430 began irradiation in November 1978; the gamma spectrometer was installed in 1979 and the data for this analysis were obtained at an average burnup of ~ 3000 MWd /t UO_2. The fuel pellets are pressed and sintered UO_2, the grain size ranging from ~ 20μm at the outer radius to ~70μm in the middle of the pellet. The fuel-cladding gap sizes are 0.1 mm and 0.23 mm. The maximum peak centerline fuel temperatures over the life of the rods have been below 1560 K; the peak fuel centerline temperatures during the fission gas release tests were ~1250 K, the bulk average fuel temperature ~ 850 K. The average linear heat ratings of the rods were 22 to 23.5 Kw/m, with peak to average of 1.2.

The isotopic release rates for Xe and Kr were measured at a steady state rod power of 26 to 28 Kw (~25 w/gm) on three different occasions during a one week period; the reactor had been at constant power for sufficient time to allow all of the Xe and Kr isotopes used in the analysis to come to equilibrium prior to the fission gas release tests. The steady state release rates were determined by flowing a constant stream of He(~1 ℓ/min), through each fuel rod and acquiring 4 to 5 measurements of the content of the gas stream using on line gamma spectroscopy.

The technique described by Carroll[1] was used to measure the iodine release rate. The ^{135}Xe release measured at steady state power consists of ^{135}Xe released from the UO_2 in its gaseous state and ^{135}Xe coming from the decay of ^{135}I plated out on the surface of the cladding and piping. A stable ^{135}Xe release rate indicates that both the release of ^{135}Xe from the fuel as a gas and the release of ^{135}Xe as a result of ^{135}I decay are at equilibrium. At this stage the amount of ^{135}I being released from the fuel and plating out is equal to that decaying to ^{135}Xe. If the reactor is scrammed the production and release of ^{135}Xe and ^{135}I from the UO_2 essentially stops, and the ^{135}Xe measured after scram is a result of the decay of the plated out ^{135}I. Thus, by measuring the release of ^{135}Xe after scram the equilibrium ^{135}I release rate can be determined.

NOBLE GAS RELEASE

The Xe and Kr fission gases swept out of the fuel rods were measured in a continuous manner by gamma-spectrometry. The release rate, R, was calculated for each isotope and used with the calculated birth rate, B, to determine the release-to-birth ratio $(R/B)_o$. The birth rates were calculated with the ORIGEN isotope generation and depletion code.[5] The mean values of $(R/B)_o$ (observed release to birth ratio) along with the standard deviation for each isotope are presented in Table 1 for each fuel rod. The difference in $(R/B)_o$ between the two rods is not presently understood.

TABLE 1 MEASURED (R/B)o AND STANDARD DEVIATION (s) FOR Xe AND Kr ISOTOPES

	0.1-mm-Gap Rod		0.23-mm-Gap Rod		Mean
Isotope	$(R/B)o \cdot 10^5$	$s \cdot 10^5$	$(R/B)o \cdot 10^5$	$s \cdot 10^5$	$(R/B)m \cdot 10^5$
^{139}Xe	5.1	0.3	2.8	0.1	4.0
^{137}Xe	8.4	0.1	2.8	0.1	5.6
^{138}Xe	10.4	0.2	3.5	0.1	7.0
^{135m}Xe	15.4	0.7	9.9	0.6	12.7
^{135}Xe	45.4	5.0	30.7	6.0	38.1
^{90}Kr	5.5	0.4	3.5	0.1	4.5
^{89}Kr	9.5	0.3	3.2	0.1	6.4
^{87}Kr	22.4	1.2	9.9	0.7	16.2
^{88}Kr	43.3	2.0	16.2	1.7	29.8
^{85m}Kr	28.8	3.2	14.2	1.1	21.5

To compare the R/B ratios from the full size fuel pellets with previously published results, the R/B ratios must be normalized by the specimen surface-to-volume (S/V) ratio. The true surface area of the spheres has been determined[6] to be up to three times the geometric surface area, and that of the pellets up to 10 times the geometric surface area.[6] The 0.1-mm-gap rod IFA-430 fuel pellets are 12.7 mm long with a radius of 5.405 mm resulting in a geometric surface-to-volume ratio of 0.53 mm^{-1}. In Figure 2 the IFA-430 results are compared to Friskney and Turnbull's[4] data on 1.2-mm-diameter spheres. Figure 2 shows that the R/B ratios for the small spheres and the IFA-430 pellets are of the same order of magnitude, the data for the spheres generally falling within the range of the pellet data.

The mechanisms for release of the fission gases from the UO_2 are recoil, knockout, and diffusion. Recoil and knockout are usually the dominant mode of release at temperatures below 1000 K, and diffusion begins to dominate at temperatures above 1000 K. Because the fuel temperature range in IFA-430 is from ~750 K (at the fuel outer surface) up to ~1250 K (at the centerline), the release of fission gases is expected to be a combination of all three mechanisms. First the knockout and recoil mechanisms will be discussed and then, using experimentally determined diffusion coeffecients, the possibility of describing the release as a diffusion process will be discussed.

The parameter most often used to investigate release mechanisms is the decay constant (λ) dependence of the R/B ratios. The mean values of $(R/B)_o$

for the 0.1-mm-gap rod are plotted as a function of the decay constant in Figure 3. The solid line drawn through the data is the least-square-fit line, which has a slope of -0.33. Olander[7] has shown that theoretically the R/B ratio for recoil and knockout release can be expressed as

$$\frac{R}{B} = \frac{1}{4}\eta\,\frac{Sg}{V}\,\mu + \frac{1}{4}\,\frac{St}{V}\,\frac{\mu\alpha F}{N\lambda} \qquad (1)$$

where

η = fraction of direct recoils not embedded in fuel or cladding
St = total surface area of fuel
Sg = geometric surface area of fuel
μ = fission fragment range
N = uranium atom density
α = knock-on ejection yield
V = fuel volume
λ = decay constant
F = fission density

The first term on the right is the recoil contribution, and the second term is the knockout contribution. Equation (1) indicates that if knockout is the dominant mechanism R/B should show a λ^{-1} dependence and if recoil is the dominant mechanism R/B should show no λ dependence; however, the measured data show a $\lambda^{-0.33}$ dependence suggesting that there is a contribution from both recoil and knockout. These results are in agreement with the results of Soulhier[8], which show a $\lambda^{-0.2}$ to $\lambda^{-0.3}$ dependence for 95% dense sintered pellets at temperatures of ~500 K (note that knockout and recoil are independent of temperature). However, since the temperature range for IFA-430 is 750 K to 1250 K there may also be some diffusion release.

At fuel temperatures in the range 1000 to 1200 K Friskney and Turnbull's [4] data show only a very weak dependence of R/B (and thus the diffusion coefficient D) on temperature. The R/B ratio can be expressed as[4]

$$R/B = 3 \; \frac{\coth y - 1/y}{y(1 - y^2/x^2)} + \frac{\coth x - 1/x}{x\,(1 - x^2/y^2)}\;, \qquad (2)$$

where $x = a\sqrt{\lambda_1 D_1}$, $y = a\sqrt{\lambda_2 D_2}$, a = effective sphere size; D_1, D_2 and λ_1, λ_2 are the diffusion coefficients and decay constants for the precursor and the noble gas, respectively. This expression takes into account diffusion of the noble gas precursor. Noting the dependence of Equation (2) on λ, for fuel average bulk temperatures of ~850 K, if the measured R/B shows an approximate $\lambda^{-0.5}$ dependence then the release should be describable as a diffusion process. Thus, from the present data, the dominant release mechanism cannot be conclusively determined, and is expected to be a combination of recoil, knockout, and diffusion mechanisms.

Applying Equation (2) and Friskney's diffusion coefficients, D_1 and D_2, extrapolated to the average IFA-430 fuel temperatures, the calculated release-to-birth ratios (R/B)c, are compared to the measured, (R/B)m, ratios in Table 2. Note that these diffusion coeffecients are experimentally determined and implicitely contain the recoil and knockout contribution to R/B. An effective sphere radius of 100 μm was used, based on a = 3V/St; St being the total surface area for diffusion as given by Belle.[9] Table 2 shows that apparently the release of Xe and Kr from full size UO_2 pellets can be estimated using Equation (2). This conclusion is based on only the IFA-430 data which is in a relatively low temperature range (750-1250 K) and may not be applicable to high temperature (>1250 K) release. This is a

TABLE 2. CALCULATED, (R/B)c, AND MEASURED, (R/B)[a]m, RELEASE TO BIRTH RATIOS

Isotope	^{138}Xe	^{135m}Xe	^{135}Xe	^{87}Kr	^{88}Kr	^{85m}Kr
(R/B)m	7.0 E-5	1.3 E-4	3.8 E-4	1.6 E-4	3.0 E-4	2.2 E-4
(R/B)c	3.6 E-5	1.1 E-4	1.6 E-4	1.1 E-4	1.1 E-4	2.0 E-4
$\frac{(R/B)m}{(R/B)c}$	1.9	1.2	2.4	1.5	2.7	1.1

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[a] See Table 1.

preliminary analysis and further investigation is needed to understand the differences between the calculated and measured R/B for ^{138}Xe, ^{135}Xe, and ^{88}Kr.

IODINE RELEASE

The release rate of ^{135}I was determined by scramming the reactor and measuring the ^{135}Xe daughter of ^{135}I for 20 hours following scram. Figure 4 shows the normalized release rate of ^{135}Xe prior to and following the reactor scram. The ^{135}Xe release rate was essentially constant for the three days prior to scram indicating that the ^{135}Xe and ^{135}I release rate had come to equilibrium. The ^{135}Xe release rate after scram drops off with the 6.6 hour half-life of its ^{135}I precursor; by extrapolating this decay back to reactor scram the equilibrium ^{135}I release rate at power is determined. Figure 4 shows that, at equilibrium conditions just prior to scram the total ^{135}Xe release rate consisted of 52% from the decay of plated out ^{135}I and 48% from direct release of ^{135}Xe as a gas from the UO_2. Thus, the ^{135}I release rate is 0.52 times the measured ^{135}Xe total release rate, which results in a R/B rate for ^{135}I of 2.52 x 10^{-4}. The diffusion coefficients for iodine given by Friskney and Turnbull[4] only go down to temperatures of ~1000 K and, as they are nonlinearly dependent on temperature, cannot be extrapolated to the fuel temperatures in IFA-430 (~850 K bulk average). Thus, a meaningful comparison of calculated and measured results cannot be made at present.

CONCLUSIONS

The preliminary results from the IFA-430 experiment, have shown that measurement of the release of short lived Xe, Kr and I fission products during nuclear operation of LWR type fuel is possible. The release at bulk average fuel temperatures of ~850 K appears to be due to a combination of recoil, knockout, and diffusion mechanisms. The R/B ratios are in the range observed in small sphere specimen experiments, when corrected for the specimen S/V ratio. The diffusion equations that take precursor mobility into account, coupled with the diffusion coefficients determined by Friskney and Turnbull from small specimen experiments, appear to predict, within a factor of about two, the release of Xe and Kr from the IFA-430 pellet fuel.

The strong temperature dependence of the available Iodine diffusion coefficients, and the current absence of reliable diffusion coefficients in the IFA-430 fuel temperature range have precluded comparison of the IFA-430 iodine release data with calculated results. However, the release of ^{135}I has been shown to be the same order as the ^{135}Xe release.

ACKNOWLEDGMENT

The authors thank R. W. Miller and the Halden Project Staff for their support and cooperation, and W. Olson for performing the ORIGEN calculations.

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NOTICE

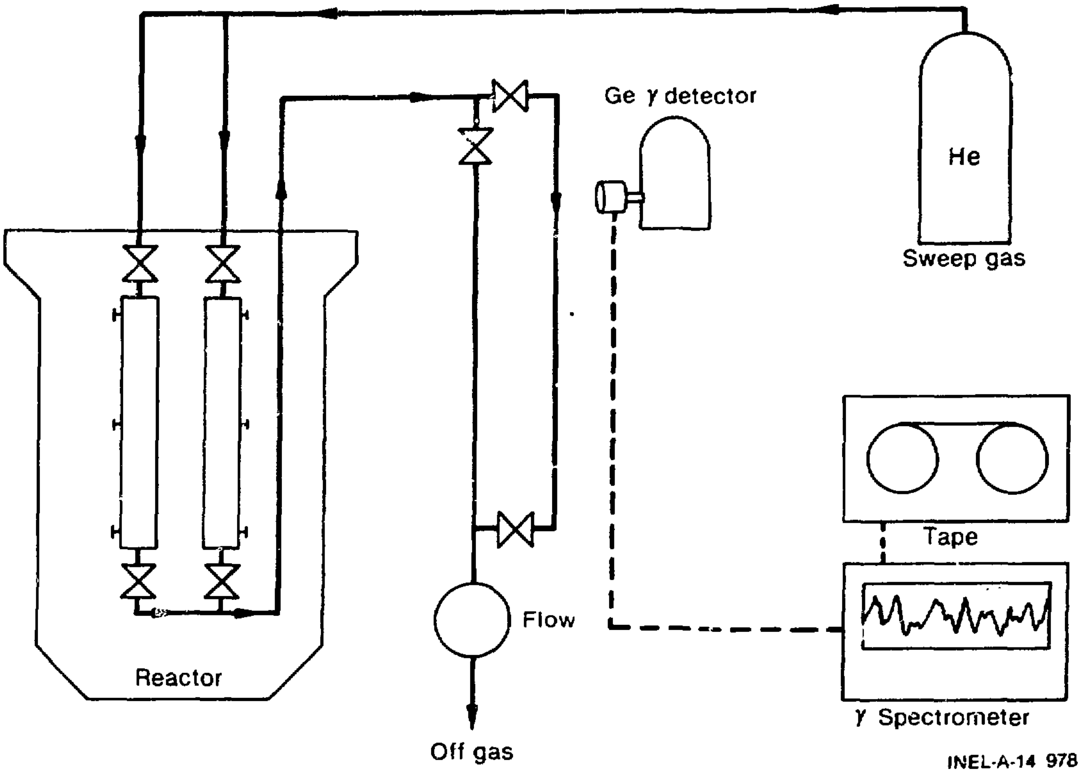

Figure 1. IFA-430 sweep gas and fission product measurement system.

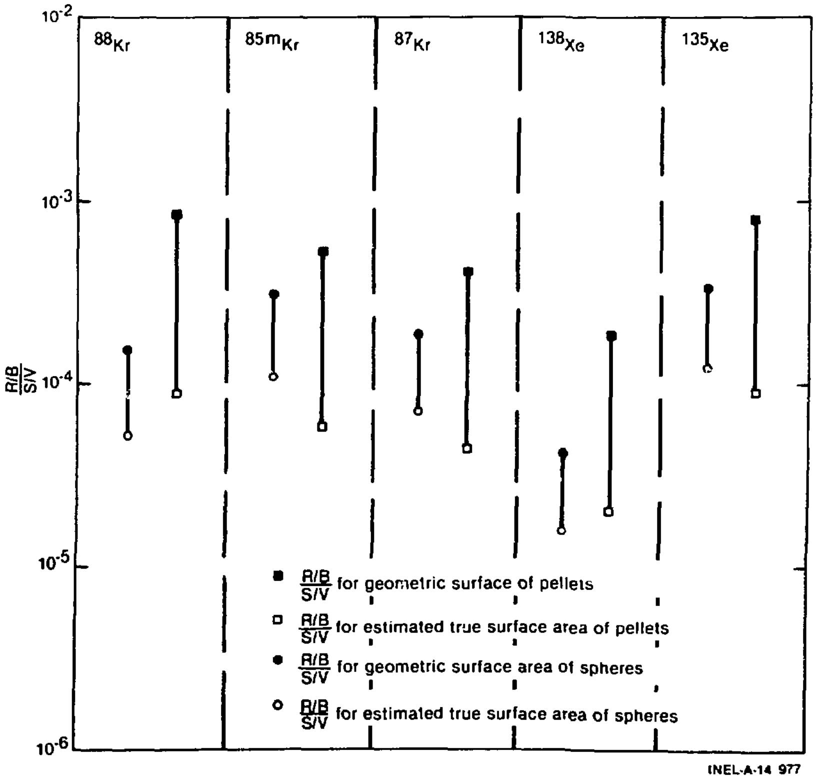

Figure 2. Comparison of the Release-to-Birth ratio, normalized by the surface-to-volume ratio, of the IFA-430 fuel pellets (■) and 1.2 mm diameter spheres[4] (●).

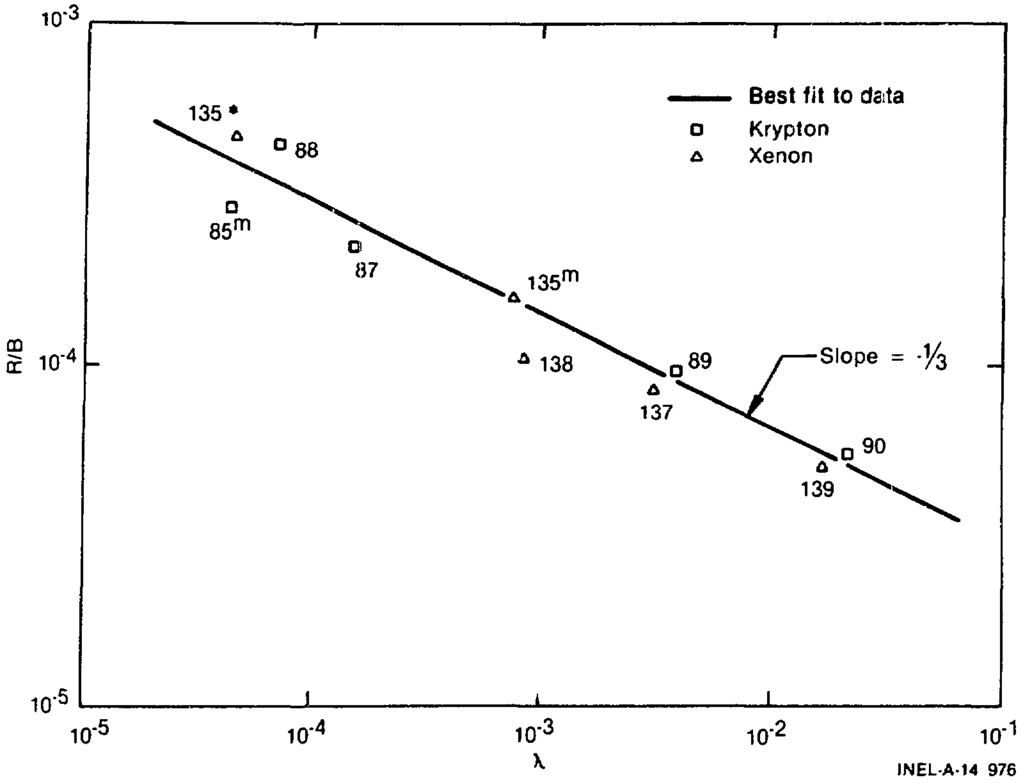

Figure 3. Measured R/B ratios as a function of decay constant ($*^{135}Xe$ corrected for neutron capture).

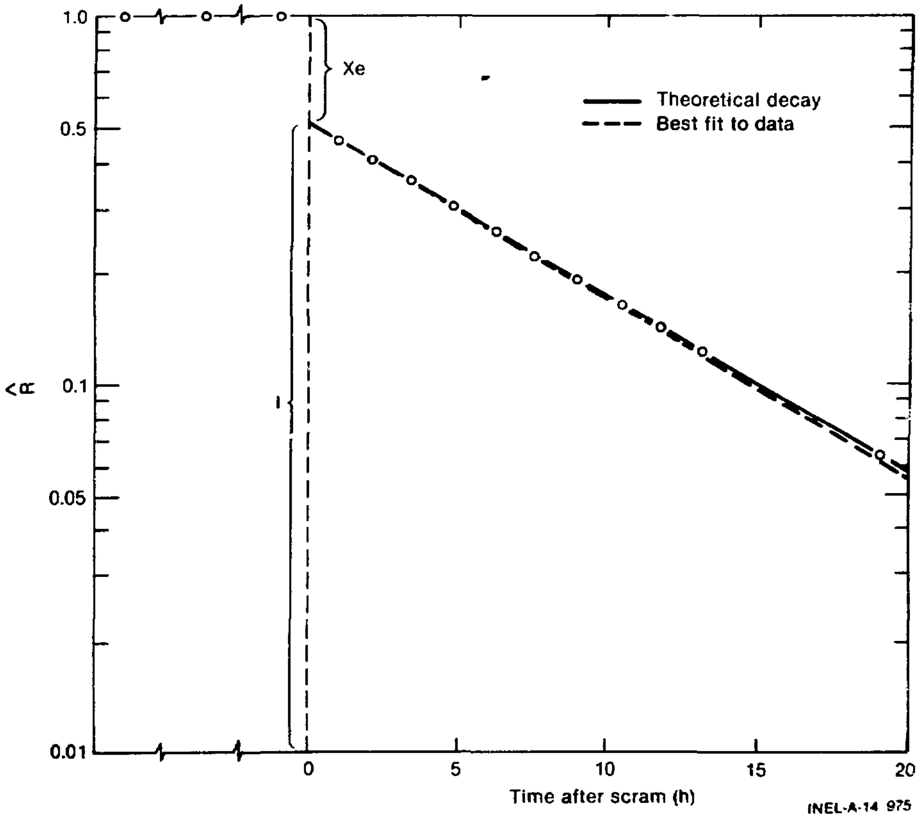

Figure 4. Normalized 135 Xe release rate (R) as a function of time for three days prior to scram and following scram indicating ^{135}I release rate is ∿ 50% of ^{135}Xe release rate.

IODINE SCRUBBING IN STEAM GENERATOR TUBE RUPTURE ACCIDENTS

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ABSTRACT

A computer code called WISE has been written to analyze iodine behavior in SGTR accidents. The code has been used to investigate the effects of water droplets which may be entrained in steam bubbles on iodine removal by the secondary water. Additionally, a sensitivity study has been performed to determine which variables have the greatest impact on iodine behavior. The iodine mass transfer coefficient from steam and the bubble rise velocity were found to have the greatest effect on iodine removal by the secondary water. Bubble moisture fraction was more important than droplet size. Changing the bubble moisture fraction from 0.83 to 0 for bubble rise heights of 0.25 and 5 m resulted in an increase in the iodine decontamination factor from 0.8 to 1.0 and from 0.09 to 0.9, respectively.

INTRODUCTION

Nuclear power plant licensing in the U.S. requires an evaluation of the potential radiological consequences resulting from a PWR steam generator tube rupture (SGTR) [1]. In such accidents the pressure differential between the primary and secondary coolants of ~9 MPa causes flow of primary coolant to the secondary side of the steam generator where part of it flashes to steam forming steam bubbles in the secondary water. These bubbles carry iodine volatilized from the primary water. Scrubbing of the iodine will occur as the bubbles rise through the secondary water. Some of the iodine released to the secondary system will transport to the environment, either through the safety relief valves for an accident which involves coincident loss of offsite power (and thus the prevention of steam dump to the condenser) or through the condenser air ejector when offsite power is maintained. The magnitude of the iodine environmental source term depends on the removal processes which operate on the iodine as it transports through the secondary system.

Postma and Tam [2] have suggested that some of the leaked primary coolant could be fragmented into small droplets which would be suspended in the steam bubbles. Iodine removal from these droplets by secondary water scrubbing may be much less efficient than scrubbing from steam bubbles. Consequently these water droplets may have a major impact on SGTR accident analyses. The purpose of this paper is to investigate the impact of these water droplets on iodine removal. Additionally, the relative importance of variables which affect iodine removal has been studied. This research was accomplished by writing a computer code called WISE to analyze iodine removal and using it to perform a sensitivity study.

THE WISE CODE

WISE (Washout of Iodine from StEam) is a Battelle-developed computer code which analyzes the removal of iodine from steam bubbles containing entrained water droplets when they pass through a pool of water. WISE accounts for iodine mass transfer between water droplets and steam in the bubble and between bubble steam and bulk water. The steam bubble and its entrained water droplets are assumed to be in thermodynamic equilibrium with the secondary coolant. The bubble size during its rise through the boiler water, and the number and size of the entrained water droplets are also assumed to remain constant.

The key input variables to WISE are iodine mass transfer coefficients from steam and water, steam bubble size and rise velocity, water droplet size, bubble moisture fraction (i.e., the fraction of bubble water mass present as droplets), bubble rise height, primary coolant leak rate and iodine concentration, and the thermal-hydraulic conditions in the boiler water. The principal output from WISE is a set of dimensionless numbers which represent various iodine removal efficiencies. Three are used in this paper. They are the overall iodine decontamination factor (DF), the droplet scrubbing factor (SF), and a partition factor (PF). These quantities are defined as:

DF = mass I_2 in secondary water/mass I_2 released from primary system

SF = mass I_2 in droplets at end of scrubbing/mass I_2 in droplets at beginning of scrubbing

PF = mass I_2 in droplets released from steam generator/mass I_2 released from steam generator.

Other computer codes have been written to calculate iodine decontamination factors for SGTR accidents, most notably DEFACT [3]. These earlier codes do not account for the presence of water droplets in the steam bubble and thus WISE represents an extension of such codes.

SENSITIVITY ANALYSIS

A sensitivity analysis was performed with the WISE code to investigate the relative importance of variables which affect iodine transport and removal and, in particular, to study the effect of the presence of water droplets on iodine removal. Geometries and thermal-hydraulic conditions appropriate for a tube rupture accident in a steam generator with loss of offsite power were used [3]. Variables included in the sensitivity analysis were:

(1) Iodine mass transfer coefficient from steam, K_1

(2) Iodine mass transfer coefficient from water, K_2

(3) Steam bubble radius, R_1

(4) Bubble rise velocity, V_B

(5) Water droplet radius, R_2

(6) Bubble moisture fraction, F_M

All other code input variables were held fixed at the values given in Table I. The analysis of the accident was performed for 720 seconds of accident time.

Iodine transport and removal mechanisms can be expected to depend sensitively on the physicochemical forms of iodine which exist in the system. Some possible forms are: inorganic (I_2 and HOI), particulate, and organic (e.g., CH_3I). Investigations have been made of the physicochemical form of iodine in primary and secondary water and chemical reactions in the secondary system during SGTR accidents have been analyzed [2]. It is believed that iodine in the primary coolant would be mainly elemental in form and that when it is released to the secondary side only a small fraction (<1%) is converted to methyl iodide [2]. Consequently, for the purposes of the present work, it has been assumed that the iodine is in elemental form.

A base case run was made with WISE in which all of these input variables were set at the best-estimate values given in Table II. Also shown in Table II are low and high values of each of the variables. All these variable values were selected on the basis of engineering judgment supported by available data. The high and low values are intended to span 95% of the variable distributions. These ranges reflect uncertainty in our knowledge of the values of these variables. They were used to make two runs of WISE for each variable with that variable set at its high and low values, respectively, and all other variables set at their best-estimate values. This procedure constitutes running a one-at-a-time statistical design. It provides first-order sensitivities but does not give any information about interactions between variables. The matrix of WISE code runs made is given in Table III.

Results of the sensitivity analysis are shown in Table IV. The decontamination factors given in this table are displayed graphically in Figure 1 where they are plotted against normalized variable values. The high and low values are represented by +1 and -1, respectively. From Table IV and Figure 1, it can be seen that the variables which have the greatest effect on iodine removal by the pool are the iodine mass transfer coefficient from steam and the bubble rise velocity. Of approximately comparable importance are the bubble size and moisture fraction. Droplet size and the iodine mass transfer coefficient from droplets appear to be unimportant within the limitations of the transport models in WISE. It is interesting to note that while calculated droplet scrubbing factors (Table IV) range from 9×10^{-4} to 0.98, the fraction of the total iodine released from the steam generator in water droplets (the partition factor, PF) is close to 1.0 for all cases. For comparative purposes, the WISE code was run for a zero bubble moisture fraction (Run 14 of Table IV) and the iodine decontamination factor was found to be 1.0. This compares to a value of 0.8 when the moisture fraction is 0.83. Clearly, the presence of water droplets has a marked effect on the calculated decontamination factors.

The results of Table IV and Figure 1 were calculated for a bubble rise height of 5 m. This represents an average value for a steam generator tube approximately 9 m high. A sensitivity analysis similar to that described here was also performed for a rise height of 0.25 m. The relative sensitivities of the variables were the same at both values of bubble rise height. Naturally, calculated decontamination factors were substantially lower for the lower rise height. For example, the base case runs with and without water droplets present gave decontamination factors of 0.09 and 0.9, respectively. These may be compared to values of 0.8 and 1.0 for a rise height of 5 m. Again, the importance of the presence of water droplets is demonstrated.

CONCLUSIONS

Several conclusions may be drawn from these results. In the case of iodine mass transfer, the transport of iodine from steam bubbles to secondary water is most important. For bubble dynamics, both bubble size and rise velocity are important while for droplet dynamics, the moisture fraction appears to be more important than droplet size. These conclusions provide useful guidance to further analytical and experimental research to resolve uncertainties in the calculation of iodine environmental releases during SGTR accidents. It should be noted that these conclusions depend upon the validity of the models in the WISE code, and to some extent on the accuracy of the best estimate values and ranges of the variables used in the sensitivity analysis. Nevertheless, since the code employs fundamental mass transfer equations to describe iodine transport and comparisons of decontamination factors are made on a relative rather than absolute basis, a reasonable measure of confidence can be placed in these conclusions.

Battelle's Columbus Laboratories is currently performing a project for the Nuclear Regulatory Commission to develop a computer code to analyze SGTR accidents which will incorporate calculations of iodine transport and steam generator thermal hydraulics. This new code will incorporate some potentially important processes not included in WISE such as droplet loss from bubbles and droplet removal by moisture separators.

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TABLE I. WISE CODE INPUT VARIABLES HELD FIXED DURING SENSITIVITY ANALYSIS

Time (s)	Steam Flow Into and Out of Steam Space (kg/s)	Leak Rate From Primary System (kg/s)	Mass of Secondary Water Near Leak (10^4 kg)	Mass of Steam in Steam Space (10^3 kg)
0	1070	0.06	7.43	7.04
60	37.8	19.5	7.00	8.87
120	37.8	17.7	7.06	8.73
240	24.5	17.3	7.31	8.52
360	19.2	17.0	7.55	8.36
480	18.0	16.8	7.80	8.19
600	17.5	16.4	8.04	8.03
720	17.5	15.4	8.29	7.87

I_2 concentration in primary coolant - 1000 μCi/kg

Bubble rise height - 5 m.

TABLE II. LOW, BEST-ESTIMATE, AND HIGH VARIABLE VALUES FOR SENSITIVITY ANALYSIS

Variable	Value Low	Best-Estimate	High
K_1 (cm/sec)	0.05	0.5	1.0
K_2 (cm/sec)	0.05	0.5	1.0
R_1 (cm)	0.125	0.225	0.625
V_3 (cm/sec)	15.0	60.0	150.0
R_2 (μm)	0.5	4.0	50.0
F_M	0.7	0.83	0.9

TABLE III. MATRIX OF WISE CODE RUNS FOR SENSITIVITY ANALYSIS

Run	K_1	K_2	R_1	V_B	R_2	F_M
1	0.5	0.5	0.225	60	4	0.83
2	0.5	0.05	0.225	60	4	0.83
3	0.5	1.0	0.225	60	4	0.83
4	0.05	0.5	0.225	60	4	0.83
5	1.0	0.5	0.225	60	4	0.83
6	0.5	0.5	0.125	60	4	0.83
7	0.5	0.5	0.625	60	4	0.83
8	0.5	0.5	0.225	15	4	0.83
9	0.5	0.5	0.225	150	4	0.83
10	0.5	0.5	0.225	60	0.5	0.83
11	0.5	0.5	0.225	60	50	0.83
12	0.5	0.5	0.225	60	4	0.7
13	0.5	0.5	0.225	60	4	0.9
14	0.5	-	0.225	60	-	0.

K_1	Iodine mass transfer coefficient from steam (cm/sec)
K_2	Iodine mass transfer coefficient from droplets (cm/sec)
R_1	Bubble radius (cm)
V_B	Bubble rise velocity (cm/sec)
R_2	Droplet radius (μm)
F_M	Bubble moisture fraction.

TABLE IV. SENSITIVITY STUDY RESULTS FROM WISE CODE(a)

Run	DF	SF	PF
1	0.84	0.19	0.97
2	0.81	0.23	0.97
3	0.84	0.19	0.97
4	0.17	0.98	0.97
5	0.97	0.03	0.97
6	0.96	0.05	0.97
7	0.48	0.61	0.97
8	1.00	0.0009	0.98
9	0.51	0.57	0.97
10	0.84	0.19	0.97
11	0.80	0.23	0.98
12	0.96	0.05	0.94
13	0.67	0.36	0.98
14	1.0	-	-

DF = mass I_2 in secondary water/mass I_2 released from primary system

SF = mass I_2 in droplets at end of scrubbing/mass I_2 in droplets at beginning of scrubbing

PF = mass I_2 in droplets released from steam generator/mass I_2 released from steam generator.

(a) Results for 720 seconds into accident.

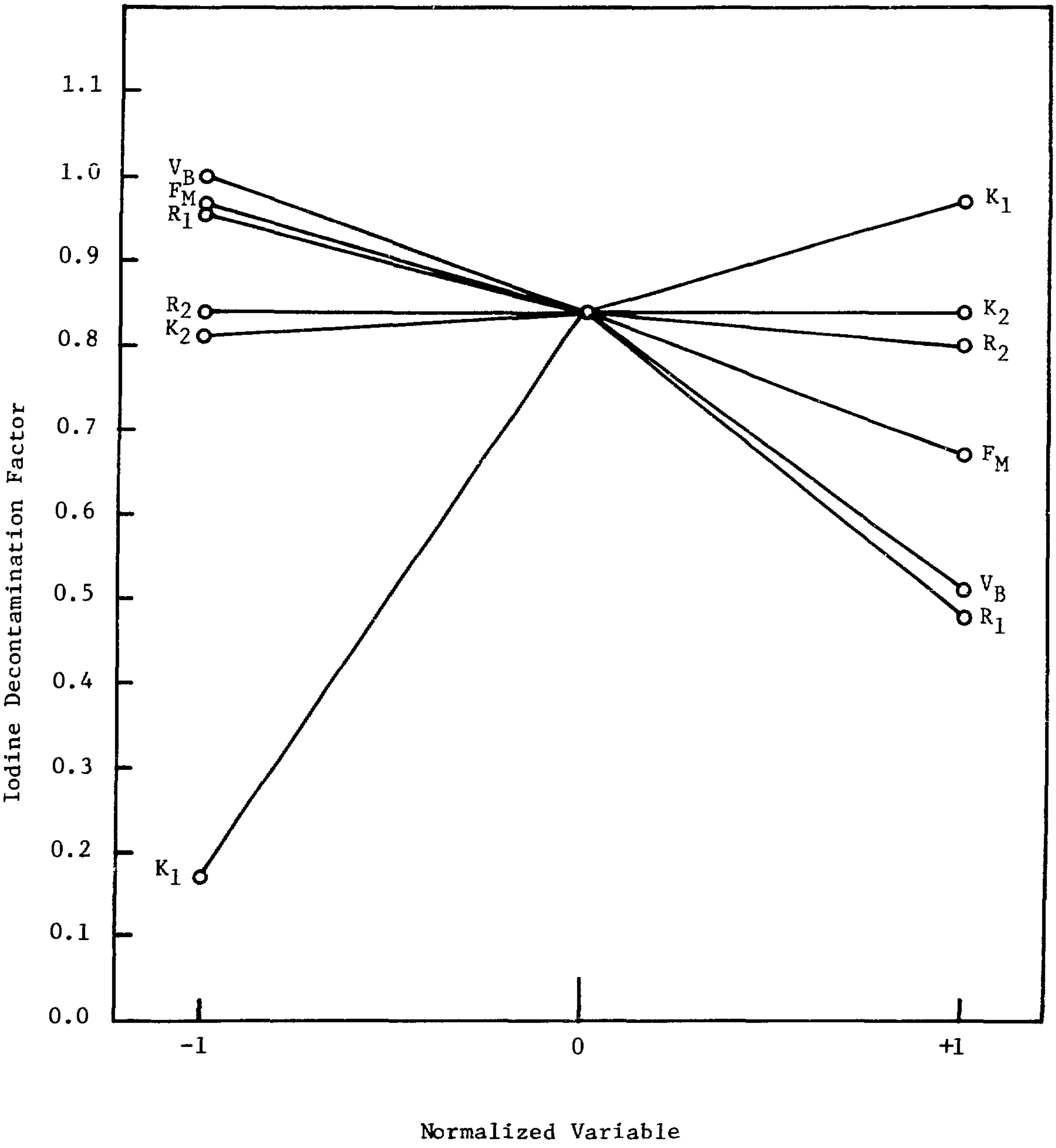

FIGURE 1. IODINE DECONTAMINATION FACTORS PLOTTED AS A FUNCTION OF NORMALIZED VARIABLE VALUES

LIQUID PATHWAYS GENERIC STUDIES; RESULTS, INTERPRETATION, AND DESIGN IMPLICATIONS

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ABSTRACT

Offshore Power Systems and the Nuclear Regulatory Commission have evaluated dose consequences resulting from a release of radioactivity to liquid pathways following a postulated core-melt accident. The objective of these studies was to compare the risks from postulated core-melt accidents for the Floating Nuclear Plant with those for a typical land-based nuclear plant. Offshore Power Systems concluded that the differences in liquid pathway risks between plant types are not significant when compared with the air pathways risks. Air pathways risk is similar to or significantly larger than liquid pathways risk depending on the accident scenario. The Nuclear Regulatory Commission judged the liquid pathways risks from the Floating Nuclear Plant to be significantly greater than the liquid pathway risks for the typical land-based plant. Although OPS disagrees with the NRC judgment, design changes dictated by the NRC are being implemented by OPS.

INTRODUCTION

As part of the review of Offshore Power Systems' (OPS) application for a license to manufacture a Floating Nuclear Plant (FNP), NRC required that studies be performed to evaluate dose consequences resulting from a release of radioactivity to liquid pathways following a postulated core-melt accident. A joint study was undertaken with NRC in which OPS performed evaluations for the FNP and NRC performed similar evaluations for land based plants (LBP). NRC then compared the results for FNPs and LBPs to determine if the liquid pathways consequences and resulting risk from core-melt accidents were less, similar or greater for FNPs. The purpose of this paper is to summarize results obtained in the OPS Liquid Pathways Generic Study for FNPs[1], compare them with results obtained by NRC[2], and discuss the respective interpretation of the results. The design requirements imposed by NRC based on their interpretation of the results[3] will be briefly described. The evolution and licensing history of the Liquid Pathways Generic Studies are discussed in another paper.[4]

ANALYSIS

For the OPS Liquid Pathways Study, two types of sources which could release radioactivity to liquid pathways were considered following a postulated core-melt accident. The larger source was that resulting from release of contaminated water collected in the containment sump to the basin in which the FNP is moored. The FNP containment is designed such that melted ice condenser water, containment spray water, safety injection water and spilled reactor coolant are collected after a LOCA on the containment floor which serves as the sump for ECCS recirculation and containment spray system recirculation.

Volatile fission products released during core melt will be washed from the containment atmosphere by containment sprays and be collected in the containment sump. This sump is isolated from the cavity region below the reactor vessel by a wier of sufficient height that overflowing of sump water into the lower reactor vessel cavity will not occur. However, it is possible for the sump water to reach the reactor vessel cavity by being pumped by the ECCS recirculation system into the reactor coolant system and out the melt hole after reactor vessel melt-through unless specific operator action is taken to secure these systems. If the plant operator takes action, ECCS recirculation is terminated thus eliminating any sump water release. OPS did analyze the case where sump release occurred and considered 15% of the sump volume as a representative estimate of the fraction of the sump water volume that might be released to the surrounding water body in the event core debris melted through the platform.

The second type of source, which generally contributes less to liquid pathways doses, is radioactivity that would be leached from core-melt debris after the debris entered the basin. To estimate the quantity of radioactivity that might be released by leaching, OPS assumed that 20% of the core debris was fragmented to small particles with the very large surface area to mass ratio of 1000 cm^2/gm (surface area approximately equivalent to 12 spheres). After an extensive review of the literature, a leach rate based on long term leach rate data with the greatest leach rate was selected. Specifically, leach rates based on leaching of Sr-89 from Nepheline - syenite glass by ground water were employed.[5] NRC in their studies employed significantly higher leach rates for core debris. The higher NRC leach rates do not have a significant effect on calculated total dose consequences because total dose consequences depend largely on the total quantity of radioactivity released. The high leach rates do however shorten the time available to effectively institute measures to confine the source to the basin (interdiction). OPS and NRC also performed calculations to bound the quantity of radioactivity that might be released by assuming all of the sump water was released soon after melt-through and that all of the core debris was extensively fragmented (to material with a surface to mass ratio of 1000 cm^2/gm) upon contact with basin water.

To calculate concentrations of radionuclides over space and time, water transport models were developed for source terms consisting of an early single release due to sump liquids and a continuous delayed release due to leaching. These calculated concentrations were then utilized in dose computations. The models conservatively assumed that the releases were point sources in the water body. No restriction on the dispersion of radioactivity by the protective barrier around the FNP was assumed.

The transport model for ocean sites was an empirical model based on work by Okubo [6]. The water concentrations are represented as a longitudinally advective, laterally dispersive plume in a model which incorporates a spatially varying dispersion coefficient. Values for the dispersion coefficient were obtained from dye tracer experiments performed at the proposed site for the Atlantic Generating Station off the New Jersey coast. A uniformly mixed water column over a constant depth of 10 meters was assumed. A 5 centimeter/second drift current was used in the model. This value is based on the annual average flow off the New Jersey coast. An examination of oceanographic data for the Atlantic and Gulf coast regions indicated that this model was appropriate for generic calculations for both regions.

For estuarine and riverine sites, the water transport methodology set forth in Regulatory Guide 1.109 [7] was employed. The estuarine model was a one-dimensional, tidally-averaged model which provided cross-sectionally averaged concentrations. The model parameters were typical of those in large east coast estuaries. The riverine model assumed a steady state flow in the river but accounted for an increase in flow downstream. The model parameters were based on the Clinch-Tennessee-Ohio-Mississippi River system.

Effects of sorption of radioactivity by both suspended and bottom sediments were taken into account for the estuarine and riverine calculations. Sediment sorption effects were evaluated by considering both scavenging by falling sediment and direct transfer from the water column to bottom sediments. For ocean sites, transfer of radioactivity to the sediment was not considered except in calculating beach exposure doses. The reason for neglecting sediment effects for ocean sites was to maximize the dose produced by the soluble activity transported by the water column.

It is significant that water transport times are about two orders of magnitude less than air transport times. Thus effective interdiction can be initiated much more easily for liquid pathways than for air pathways.

LIQUID PATHWAY RESULTS

The OPS results show that seafood ingestion is the dominant dose pathway to man for ocean and estuarine FNP sites. The next largest dose pathway is beach exposure which is a factor of 3 to 10 lower. For riverine sites, drinking water ingestion is the dominant dose pathway. Among the fission products, CS-134 is the most important dose contributor because of its relatively high biological accumulation factor, high core inventory and substantial halflife. CS-134 accounts for 73% of the total dose from seafood ingestion and 66% of the total dose from beach exposure.

Calculated results for 50-year dose commitments without interdiction (obtained by both OPS and NRC) are summarized in Table 1 for both the case which considered releases that are reasonable for a postulated core-melt accident and for a bounding case. The bounding case assumes that all the radioactivity associated with containment sump liquids is released to the basin. In addition, all of the core debris was assumed to fragment to particles with an area to mass ratio of 1000 cm^2/gm by OPS. It can be seen from Table 1 that NRC and OPS results for population dose are in substantial agreement for both cases. For the NRC likely release case, release of 50% of the sump liquids was assumed rather than 15% employed in the OPS evaluation which accounts for a significant portion of the difference in values. The sump release is the greater dose contributor since most of the Cs initially is volatilized from the debris, scrubbed to the sump liquid and would be released with sump liquid.

With the NRC assumption of rapid release of sump liquids and rapid leaching from core debris, significant quantities of the longer lived radionuclides could escape from the basin containing the FNP before the basin could be isolated by interdiction measures, particularly at estuarines sites. NRC concluded that for estuarine sites with open protective structures that allowed direct access and exchange with the surrounding waters, contamination could be persistent and long term. They further concluded that socio economic impacts of such long term contamination could be unacceptably high.

For land based plants, NRC concluded that source interdiction could be effective because of the long transport times through the ground. Evaluations for land based plant sites with potential for rapid groundwater transport were not included in the NRC analysis. Thus, the NRC assumed that source interdiction for land based plants would reduce the population dose by 3 orders of magnitude.

For the FNP, the more effective interdiction approach is pathways interdiction which can reduce dose consequences by at least 3 orders of magnitude.[1] However, the NRC assumed pathway interdiction would only be applied for those individuals whose dose commitment would be greater than 5 Rem without interdiction. As a result, dose reduction from pathways interdiction was arbitrarily limited by NRC to about a factor of 10. OPS believes such an arbitrary assumption is inappropriate and in fact interdiction would be applied to the maximum extent feasible. Thus, on the basis of effective interdiction, the calculated doses for the FNP are within a factor of 10 of those for a LBP.

COMPARISON WITH AIR PATHWAYS

Differences in liquid pathways dose consequences for FNPs and LBPs were deemed to be significant in the NRC comparison. The dose consequences for FNPs were judged by the NRC to be significantly greater particularly when interdiction as defined by NRC is assumed.[2,3] OPS did then, and continues to, disagree that the differences are significant when dose consequences via liquid pathways are compared with those via air pathways for postulated accidents beyond the design basis.

For evaluating dose consequences via air pathways, OPS employed the methods of the Reactor Safety Study[8] (WASH-1400). The seven radioactivity release categories for PWR core-melt accidents described in WASH-1400 were utilized along with the associated source terms. For LBPs, probabilities for each release category were taken directly from WASH-1400. For the FNP, the same seven release categories were utilized with their associated probabilities developed by OPS for the FNP ice condenser design. The FNP probabilities for each release category reflect design differences between the FNP and the WASH-1400 PWR land-based plant.

The assumed population distributions and meteorology in the vicinity of the FNP and LBP sites are listed in Part II of the OPS Environmental Report.[9] Three hypothetical sites were considered for each type of plant: a New Jersey Coast site, a North Carolina coast site and a Southern Florida coast site. The resultant dose distribution curve calculated for the air pathway is compared with the dose distribution curve for liquid pathways in Figure 1. From Figure 1, two conclusions are apparent. First of all, for the more probable of the postulated core melt accident scenarios, dose consequences via liquid pathways and air pathways are similar. Second, if total residual risk associated with air pathways releases is compared with total residual risk via liquid pathways, the residual risk via air pathways is substantially larger.

CONCLUSIONS

The above analysis indicates that calculated dose consequences via liquid pathways and air pathways resulting from postulated core-melt accidents differ in two very important ways which are:

1. There are no acute fatalities calculated to occur via liquid pathways.
2. With realistic interdiction methods that can be expected to be applied for liquid pathway releases, liquid pathway population dose consequences will be significantly less than those occurring via air pathways.

In the Final Environmental Statement, Part III[3], NRC concluded that calculated dose consequences in terms of total population dose for air pathways and liquid pathways were similar for the more probable core-melt scenarios and concluded this aspect of similarity was significant. They therefore request that the concrete biological shield base mat beneath the reactor vessel in the FNP be replaced with magnesium oxide or equivalent refractory material which would increase resistance to melt-through by molten core debris and which would not react with the core debris to form a large volume of gases. The NRC judged that the magnesium oxide would delay core melt through for a few days. providing increased time to institute interdictive measures to assure that released radioactivity would be contained within the basin or, for the case of ocean sites with an open breakwater, to assure that interdictive measures can be taken to minimize subsequent dose effects. Both OPS and NRC concluded that interdictive methods can be employed to reduce dose consequences via liquid pathways to very low levels.

While OPS does not believe that technical data generated in the LPGS studies support the need for adding a layer of refractory material to the design for the purpose of delaying core-melt through, we are nonetheless proceeding with the changes dictated by the NRC so that the license review can continue.

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TABLE 1
CALCULATED CORE - MELT CONSEQUENCES* VIA LIQUID PATHWAYS(1,2)

	MOST LIKELY CASE			BOUNDING CASE		
	FNP (OPS**)	FNP (NRC***)	LAND BASED PLANT	FNP (OPS)	FNP (NRC)	LAND BASED PLANT
Ocean Site (Seafood Ingestion Pathway)	8×10^5	4×10^6	2×10^5	5×10^6	9×10^6 (320)****	1×10^6 (20)
Estuarine Site (Seafood Ingestion Pathway)	2×10^6	2×10^7	5×10^6	1.3×10^7	4×10^7 (220)	3×10^7 (40)
Riverine Site (Drinking Water Pathway)	2.3×10^5	2×10^6	1×10^5	1.4×10^6	4×10^6 (30)	1×10^6 (5 - 1500)

* Data are population doses in man-rem assuming no interdiction. For comparison, the comparable lifetime dose from natural background radiation to the 1.2 million people considered for the ocean site seafood ingestion pathway is about 5×10^6 man-rem.

** Assumes release to liquid pathways of 15% of radioactive water in containment sump.

*** Assumes release to liquid pathways of 50% of radioactive water in containment sump.

**** Data in parenthesis are maximum individual doses in rem assuming 5 years exposure.

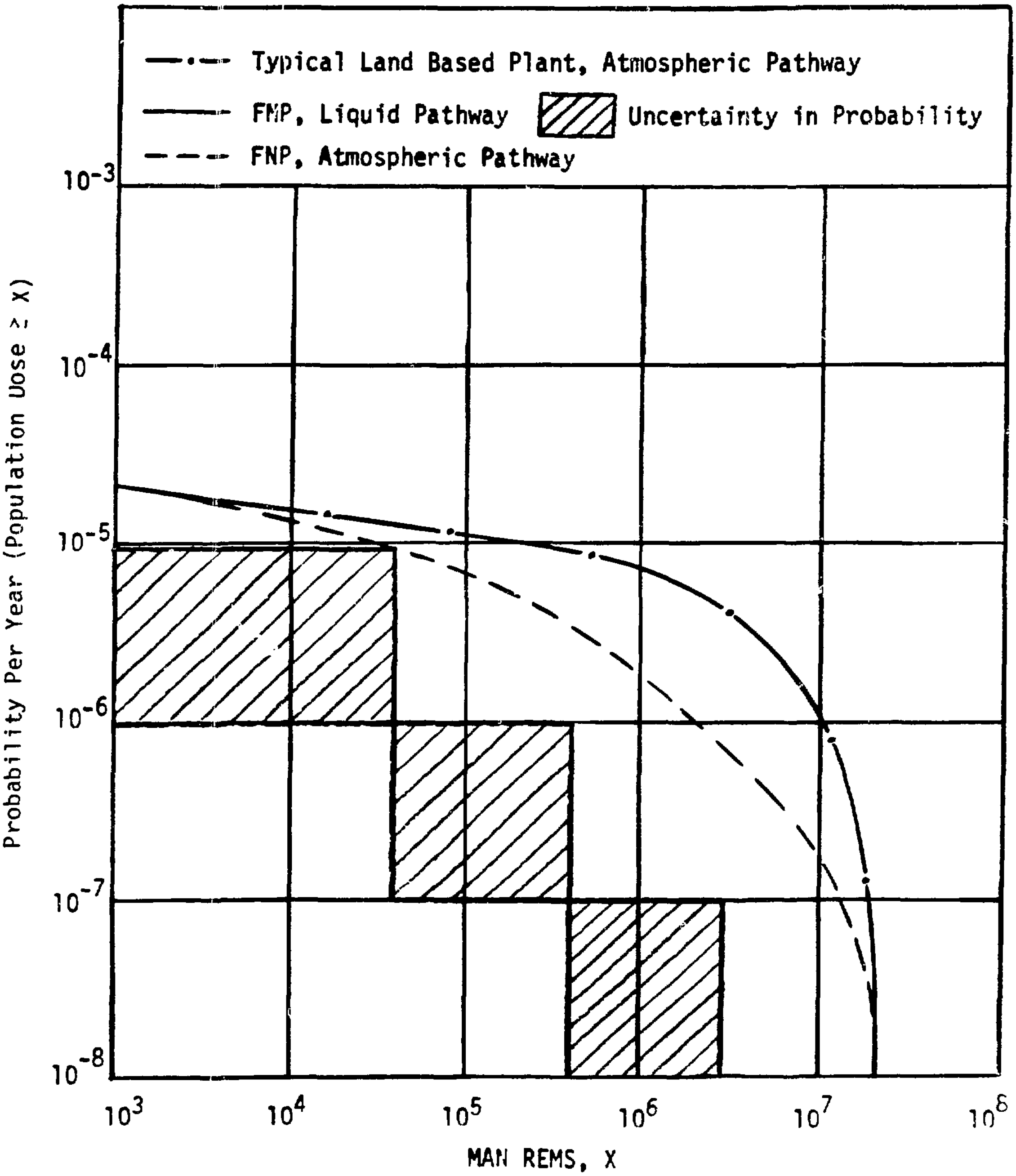

FIGURE 1 - Comparative Probability - Consequence Distribution for Liquid and Air Pathways (50 Year Dose Commitment)

AN ASSESSMENT OF LWR PRIMARY SYSTEM RADIONUCLIDE RETENTION IN MELTDOWN ACCIDENTS USING THE TRAP COMPUTER CODE

by

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ABSTRACT

The TRAP computer code has been used to determine the extent of radionuclide deposition in LWR primary systems during meltdown accidents. Uncertainty analyses have been performed to provide insight into the key variables and principal areas of uncertainty in modeling radionuclide behavior. Results of the calculations indicate that the extent of radionuclide deposition in the primary system varies widely with chemical species and accident conditions. Key contributions to uncertainty in the radionuclide deposition come from the radionuclide source term from the fuel and radionuclide deposition rates.

INTRODUCTION

The TRAP computer code has been developed by Battelle's Columbus Laboratories for the U.S. Nuclear Regulatory Commission to analyze radionuclide transport and deposition in LWR primary systems during terminated loss-of-coolant and meltdown accidents [1, 2]. It is expected that radionuclides released from the core will undergo chemical and physical changes and will deposit on various surfaces as they are transported through the primary system to the containment. It is important in safety analyses to know what fraction and what form of these released radionuclides actually reaches the containment and is available for leakage to the environment. The objective of the TRAP code is to provide this information.

Recently, a meltdown accident version of the TRAP code was completed and this paper presents the results of baseline calculations for three representative meltdown accident sequences and an uncertainty analysis for one of them. These results provide estimates of the expected radionuclide retention in the primary system and yield insight into the key variables and principal areas of uncertainty in modeling radionuclide transport.

THE TRAP CODE

The TRAP computer code models radionuclide transport and deposition in LWR primary systems during reactor meltdown and loss-of-coolant accidents. The code models interconnected compartments using a control volume approach. Radionuclide transport is superimposed on the fluid flow without coupling to it. The control volumes may be connected arbitrarily by fluid flow and a source term of radionuclides may be placed in any volume.

TRAP models the transport and deposition of both radionuclide vapors and aerosols (particulates) and accounts for phase changes of the radionuclides. Models for vapor sorption, turbulent particle deposition, laminar particle deposition, thermophoretic particle deposition, and particle agglomeration are included in the code. Radionuclide transport is modeled using the concept of a radionuclide state in which a particular radionuclide physical form is associated with a radionuclide location, e.g., iodine vapor in steam. This concept permits a flexible code design in which modeling changes and additional transport processes can easily be incorporated. Radionuclide transport can occur among the states of an individual control volume or between certain states of different control volumes if these are connected by fluid flow. The former types of transport are generally controlled by physical and chemical characteristics of the radionuclide species and are modeled using correlations for mass transfer coefficients in a system of differential equations within the code. Transport of fission products between control volumes is assumed to occur in phase with fluid transport. This transport is imposed on the code by time-dependent thermal-hydraulic data read into the code.

It is assumed that the flow system under consideration can be subdivided into a sufficient number of control volumes such that the radionuclide populaiton in each is expected to be homogeneously distributed (well mixed). Furthermore, the transport rates among states are assumed to be proportional to the amount of radionuclide in the state from which the transport occurs. Rate coefficients for transport among states are determined from correlations for vapor and particulate deposition velocities. Phase transitions of a given species are modeled mechanistically using typical mass transfer correlations.

Among the required TRAP input data are: radionuclide physical properties, primary system geometry, source term, flow connections, and thermal-hydraulics. TRAP provides as output the radionuclide masses present in each state within each control volume as a function of time. This includes the amounts of radionuclides released to the containment from the breach in the primary system.

A more detailed discussion of TRAP can be found in references [1] and [2].

BASELINE CALCULATIONS

Baseline calculations have been made for the meltdown accident sequences TC (BWR transient with failure of the reactor protection system), TMLB' (PWR transient with loss of steam generator heat sink and electric power), and AB (PWR large pipe break with loss of electric power) using best-estimate input variable values. These accident sequences were chosen on the basis of their dominant contributions to reactor meltdown risk as calculated in WASH1400 [3] and because of their representative nature. The objective was to select sequences which would span as large a part as possible of the spectrum of radionuclide deposition in the primary system. A source term which consisted of elemental iodine (I_2), cesium hydroxide (CsOH) and plutonium dioxide (PuO_2) was employed. This source term contains radionuclides of particular radiological significance and represents species with volatilities ranging from high (I_2), through medium (CsOH) to low (PuO_2). In the calculations performed iodine and plutonium dioxide always transport, respectively, as a vapor and particulate. In contrast, cesium hydroxide transports both as a vapor and a particulate. Thermal-hydraulic input data were developed using the MARCH computer code [4] augmented by hand calculations. Control volume selections were made for each accident sequence as follows.

TC: core, steam separators, steam dryers, upper vessel head, outer annulus;

TMLB': core, upper plenum, pressurizer, quench tank;

AB: core, upper plenum, lower plenum, downcomer, steam generator

Geometric data were selected to represent typical large size PWR's and BWR's.

Results of the calculations are given in Table I where the percentage is given of the radionuclides released from the fuel which are deposited in the primary system. These results indicate that retention of both I_2 and PuO_2 in the primary system is small. In contrast, retention of CsOH is substantial. The reason for this difference may be traced to the deposition mechanisms for the three radionuclides and the thermal-hydraulic conditions in the primary system for the three accidents. The mass transfer equations for CsOH are much stronger functions of temperature than the deposition velocities that govern I_2 and PuO_2 deposition. Deposition decreases with increasing temperature and this is seen in the results of Table I where the fluid and surface temperatures in the control volumes where most CsOH deposition occurs generally decrease in going from TMLB' to TC to AB.

UNCERTAINTY ANALYSES

Uncertainty analyses were performed using TRAP in order to identify the key areas of uncertainty in modeling radionuclide transport and deposition. An approach which uses statistical designs and response surfaces was employed [5]. The methodology can be explained using a simple example. Assume that the computer code is represented by the equation

$$Y = F(X_1, X_2) \qquad (1)$$

where X_1 and X_2 are input variables, Y is the output variable and F is a function which represents the computer code. In order to investigate how uncertainties which derive from our imprecise knowledge of the correct values of variables X_1 and X_2 contribute to the resulting uncertainty in the code output Y we search for a relationship of the form

$$U(Y) = U(X_1) + U(X_2) \qquad (2)$$

where U denotes uncertainty which we have not yet defined precisely. In order to accomplish this definition using a statistical quantity such as variance so that a relationship of the form of Equation (2) can be obtained it is convenient to fit a simple functional approximation to Equation (1) such as

$$Y = A_0 + A_1X_1 + A_2X_2 + A_{12}X_1X_2 \quad . \qquad (3)$$

Such an approximation is often called a response surface. It is obtained by fitting the surface to a set of runs of the computer code determined by a chosen statistical design. If variance is selected as the measure of uncertainty then Equation (3) can be used to obtain

$$V(Y) = A_1{}^2V(X_1) + A_2{}^2V(X_2) + A_{12}{}^2V(X_1)V(X_2) \qquad (4)$$

Percentage contributions of the individual terms in this expression to the total variance in the code output can then be obtained. An example is provided by

$$\frac{A_1^2 V(X_1) \times 100}{V(Y)} \quad . \qquad (5)$$

Such an uncertainty analysis was performed for the BWR accident sequence TC. Variables included in the analysis were radionuclide source term, radionuclide deposition rate (for iodine vapor and plutonium dioxide particles), vapor mass transfer coefficient for cesium hydroxide, vapor pressure, particle size, particle density, fluid flow rate, fluid temperature, and surface temperature. A statistical design was used for response surface generation which required the use of high and low values of each variable. These values were selected on the basis of engineering judgement so as to encompass approximately 95% of the variable distribution. The results of the analyses are given in Table II where percentage contributions of each input variable to the uncertainty in the amount of radionuclide deposition in the primary system are given. In the case of iodine the principal contributors are the iodine deposition velocity and surface temperature. The importance of the latter variable is due, undoubtedly, to the exponential dependence of the deposition rate on surface temperature. For CsOH the vapor mass transfer coefficient dominates and the source term from the fuel also contributes significantly. This source term is the principal contributor for PuO_2 although the fluid flow rate also makes a reasonable contribution.

It is possible to calculate a standard deviation for the primary system radionuclide deposition using the results of the code runs employed in constructing the response surface. These values are given in Table III together with the corresponding mean values. Substantial variations in the extent of radionuclide deposition are seen. The standard deviation expressed as a percentage of the mean ranges from 50% to 75%.

CONCLUSIONS

The baseline calculations which were performed indicate that deposition of radionuclides in the primary system during meltdown accidents varies widely with radionuclide chemical form and accident conditions. Results of the uncertainty analyses indicate that the key contributors to the uncertainty in primary system radionuclide deposition are the radionuclide source term from the fuel and radionuclide deposition rates. These results can be used to guide further modeling of radionuclide transport by directing attention to those areas where the greatest reduction in uncertainties can be anticipated.

An important use of TRAP will be in conditioning the radionuclides which are released from the primary system to the containment. Changes in radionuclide physical and chemical form which occur in the primary system can significantly impact the behavior of radionuclides in the containment and consequently the degree to which they will deposit there.

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TABLE I. RADIONUCLIDE DEPOSITION IN THE PRIMARY SYSTEM

	Percentage of Source Term Deposited		
	TMLB'	TC	AB
I	5.6	1.9	0.8
Cs	7.5	29.9	61.3
Pu	2.0	4.0	19.0

TABLE II. UNCERTAINTY ANALYSIS RESULTS FOR BWR ACCIDENT SEQUENCE TC

Radionuclide	Variable[a]								
	RST	RDR	MTC	VPE	MPS	PD	FFR	FT	ST
I	9	60	0	2	0	0	1	0	21
Cs	14	0	66	7	0	0	1	7	0
Pu	64	6	4	0	3	0	11	0	0

(a) RST - Radionuclide source term
RDR - Radionuclide deposition rate
MTC - Mass transfer coefficient
VPE - Vapor pressure equation
MPS - Mean particle size
PD - Particle density
FFR - Fluid flow rate
FT - Fluid temperature
ST - Surface temperature

TABLE III. MEAN VALUES[a] AND STANDARD DEVIATIONS FOR PRIMARY SYSTEM RADIONUCLIDE DEPOSITION FOR BWR ACCIDENT SEQUENCE TC

Radionuclide	Mean (μ)	Standard Deviation (σ)	($\frac{\sigma}{\mu}$ x 100)
I	0.04	0.03	75
Cs	0.29	0.14	48
Pu	0.1	0.06	60

(a) These are deposition fractions defined as the ratio of material deposited to that released.

THE NATURAL REMOVAL OF PARTICULATE RADIOACTIVITY IN AN LWR-CONTAINMENT DURING CORE MELTDOWN ACCIDENTS

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ABSTRACT

During core meltdown accidents in LWRs large quantities of fission products and core material are vaporized and released to the containment in particulate form. The behavior and the natural removal of this aerosol is calculated with the computer code NAUA. Airborne aerosol mass in the containment and leaked masses are calculated for a typical accident sequence in a German PWR.

Additionally the influence of a containment failure on the total environmental impact is evaluated as a function of time. It is demonstrated that a period of three days of containment integrity is required to minimize the aerosol mass which escapes from the containment.

INTRODUCTION

Recently an increasing interest has been observed in accident sequences in LWRs which lead to core melting. Such an event has surely not acquired a higher probability but when assessing its contribution to the risk of nuclear power a great uncertainty still exists in the prediction of the consequences. Conservatism dominates all risk evaluations to date and it is highly desirable to advance towards a more realistic description of the accident consequences.

In the overall scenario fission product behavior plays an important role, and the specific situation in meltdown accidents is characterized by two aspects: the majority of the fission products is released as solid airborne particles (aerosols) and the containment integrity might be of finite duration. Concerning containment integrity efforts are made to investigate the mechanisms which lead to a containment failure. For German PWRs the results of recent calculations [1] showed a tendency towards a longer time until a containment failure due to overpressurization. The benefit of this will become evident later in this article.

The particulate fission products are released during core meltdown together with large masses of fuel and structure material which are non-radioactive. The total aerosol mass is presently estimated to be 1...2.5 tons. Such an aerosol system is highly unstable and is effectively removed from the airborne state by mere natural processes without the help of engineered safeguards. Particulate fission products will then be removed together with the dominating inactive aerosol.

The development of an aerosol behavior code and results of recent realistic calculations will be reported in the following.

THE NAUA R&D-PROGRAM

At KfK the NAUA aerosol code has been developed to calculate the aerosol behavior in a PWR containment under typical core meltdown conditions, and to predict the airborne aerosol mass concentration inside the containment as well as leaked masses as functions of time. A scheme of the code is shown in Fig. 1.

The code has been developed on the basis of microphysical aerosol behavior equations which are well established. This immediately involves that the code is capable of accepting every accident sequence as well as every source term for particles and steam. The mathematical methods of the code [2] will not be discussed here, only the application and recent result of the calculations will be presented.

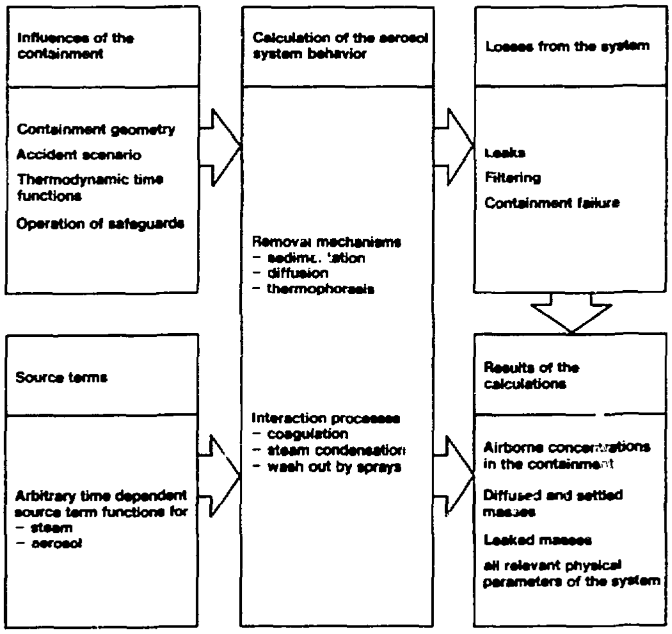

Fig. 1: Scheme of the NAUA code

The validity of this code strongly depends on the reliability of the aerosol behavior equations in its central part. These equations contain parameters which were unknown at the beginning of the code development. Therefore, an experimental program is being conducted to measure the values of unknown process coefficients, especially those related to water vapor condensation on the aerosol particles and on the walls. To date the measurement of condensation on particles has been completed [3] . The experiments on wall condensation and latent heat transfer dynamics will continue.

THE AEROSOL SOURCE FUNCTION

The calculations which are presented in the following have been performed for a typical German PWR (Biblis B) using a realistic scenario. For long term aerosol behavior calculations the containment can be considered as a single volume of 72000 m^3. The total surface was assessed as 31000 m^2 and the floor area as 2500 m^2 [4] . The latter value is rather conservative as will be discussed later. The temperature functions and steam release rates were also taken from [4] .

The releases for aerosols are defined as shown in Fig. 2. The first significant aerosol release occurs during the evaporation of the residual water in the reactor pressure vessel (RPV). The release rate is taken as linearly increasing as the core heats up and starts to melt. 80% of the total released mass is assumed to originate from this period. When the RPV is dry after 5200 sec the release becomes zero because no transport of the aerosol into the containment exists. Only when the RPV fails a second release period occurs which comprises the aerosol content in the RPV and the release from the melt as it comes into contact with the concrete structures. The gases originating from concrete decomposition serve as a carrier for the aerosol into the containment. This release is assumed to operate from 7000 to 8200 sec at constant rate. The duration is comparatively short because the melt is rapidly cooled due to the melting of the concrete and the progressive dilution with the molten concrete [1]. During this first contact of the melt with the concrete an additional amount of 110 kg of concrete aerosols is released.

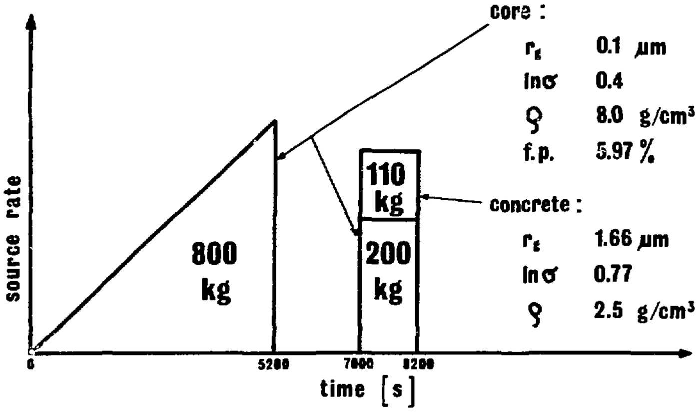

Fig. 2: Aerosol release scheme for a 1000 kg release

The particle sizes are different for core material and concrete aerosols and their best estimate values are also shown in Fig. 2, where the mean particle radius r and the variance σ are given. It should be noted that the NAUA code requires these informations about duration and particle size distributions as an input, but that the dependence of the results on these values is rather small [5] as long as the releases occur in the first few hours of the accident sequence. The total released mass is the important parameter which influences the aerosol behavior and the leaked masses.

The total released mass in Fig. 2 is 1000 kg. Parametric calculations have been done for other releases also. In this paper results for 1000 kg and 2500 kg will be presented. The best estimate value lies somewhat in between for a meltdown accident in a PWR.

The fission product content of the core material is 5.97%. This value is very important because the radiological consequences depend on the amount of radioactivity leaked to the environment, whereas the aerosol behavior depends only upon the physical properties particle size and concentration which are dominated by the inactive fuel and structural materials. Therefore, all the following results will be given in terms of aerosol mass and must be weighted by the fission product fraction when the radioactive potential is required.

AEROSOL REMOVAL AND LEAKAGE

As a first result in Fig. 3 and 4 the total airborne mass in the containment is shown as a function of time for a 1000 kg and a 2500 kg release. It should be noted that the condensation of steam on particles has not been taken into account. To our present knowledge this effect does not occur during the long time interval from RPV failure until sump water ingression. For the other phases of the accident steam condensation is such a strong effect that it should be calculated only when a reliable data base exists, which is not yet the case. As soon as we will have completed our steam transport experiments, we will include steam condensation in our calculations.

The decay of the airborne mass depends on the initially released mass in such a way that during the first day the high release decays faster than the low release. As mentioned in the previous section the floor area was taken from [4] to use a consistent set of data in spite of the fact that it is unrealistically small. Increasing the floor area would improve the decay of the mass curves in Fig. 3 and 4.

The fact that higher releases are removed faster than low releases is a well known property of aerosols and can be predicted only by an aerosol code which calculates the removal mechanisms on a microphysical basis.

The leaked masses for these two cases with 1000 and 2500 kg are also shown in Fig. 3 and 4. The parameter is the leak rate of 0.25%/d (layout value) and 1%/d respectively. In these calculations the retention of aerosols in the annular gap was neglected.

It is evident that almost the total mass leaked to the environment accumulates during the first half day. This is due to the fact that the leaked mass is, of course, proportional to the airborne mass in the containment. As the airborne mass is removed rapidly also no further significant contribution is added to the leaked mass. Further the leaked mass is proportional to the leak rate, which is also self explaining.

Finally - and this is important - the leaked mass is not proportional to the initially released airborne mass, because of the faster removal of the higher airborne mass. This means that the total leaked mass does not linearly increase with the aerosol mass released from the core. So in this example, though the total released masses differ by a factor of 2.5 (2500 kg/1000 kg) the total leaked masses differ by a factor of 1.8 (1.44 kg/0.81 kg for the 0.25% curves) only.

It must again be emphasized that the relation between the ratios 2.5 and 1.8 depends on the case considered and will be different when comparing e.g. a 1000 kg release to a 400 kg release. But a mitigation of the increase in leaked masses compared to the increase in released masses is generally observed.

OVERALL AEROSOL IMPACT TO THE ENVIRONMENT

So far we have considered only leaked masses from an intact containment. The question arises about the contribution of a containment failure to the environmental impact. From Fig. 3 and 4 it can be seen that the leaked mass

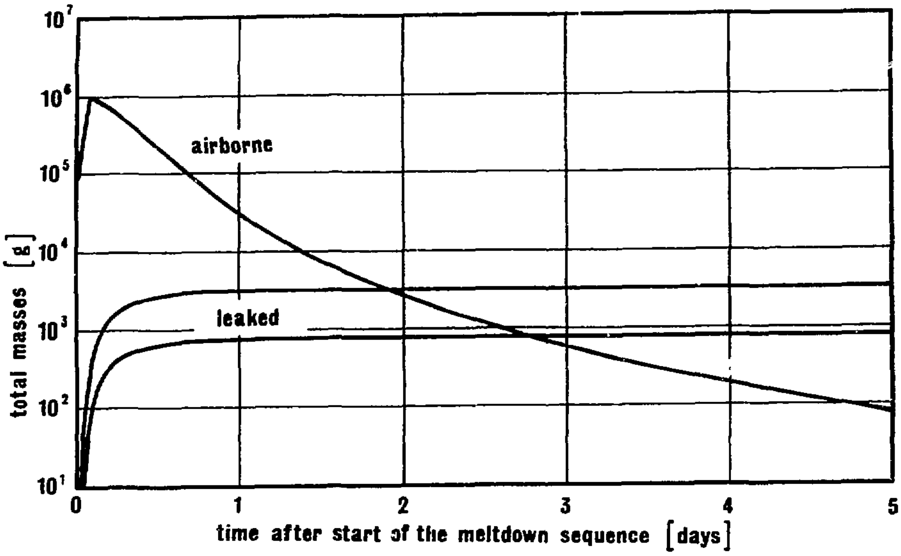

Fig. 3: Airborne and leaked masses for the 1000 kg release case Leak rates 0.25%/d and 1%/d

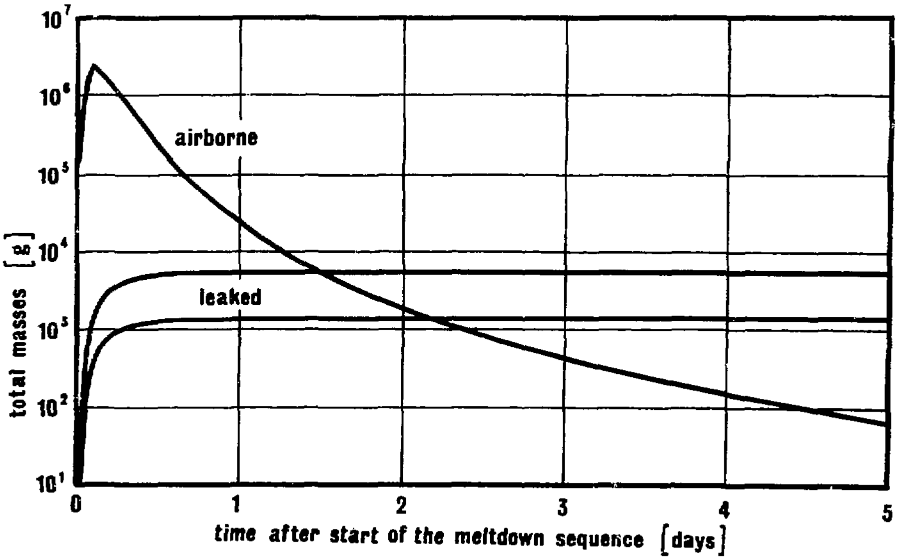

Fig. 4: Airborne and leaked masses for the 2500 kg release case Leak rates 0.25%/d and 1%/d

reaches its final value already after about half a day. At that time the airborne mass inside the containment is still orders of magnitude higher and would dominate the consequences of the accident in a case of early containment failure. Because the failure overpressure of the containment is around 9 Bar the additional mass which escapes to the environment can be considered as almost equal to the airborne inventory. It is also obvious that the mass which escapes when the containment fails is very sensitive to the timing of this event. To facilitate these considerations and to determine the period during which the containment should not fail we defined the overall "escaped mass" as the sum of the toal leaked mass and the total airborne mass in the containment. This is represented in Fig. 5 for the 1000 kg-release case with a 1%/d leak rate. The "escaped mass" curve gives the total escaped mass as a function of the time of containment failure. Now it can be seen, that the situation does not relax until three days after the start of the accident.

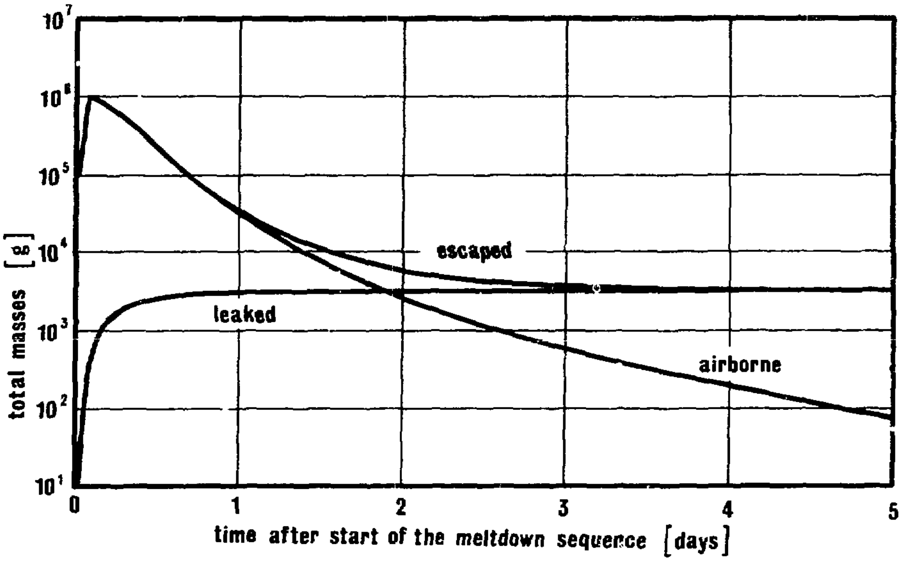

Fig. 5: Total escaped mass including containment failure

At that time the airborne mass inside the containment has been removed to such low values that a containment failure will no more contribute to the escaped aerosol mass. As was shown in [1] a containment failure will not be likely to occur before that time, steam explosions excepted.

It should be remembered at this point, that the NAUA calculations only consider particulate fission product species. Noble gases and gaseous fission products are not covered and will also not undergo efficient natural removal processes.

In Fig. 6 the total escaped mass curves for the 2500 kg- and 1000 kg-release cases with 1%/d leak rate are shown together. In close detail they show that due to the complicated nature of aerosol behavior neither curve can be considered as 'conservative' in the common meaning of this word. Depending on time either one may be higher. This again shows the necessity of calculations with the complete aerosol code.

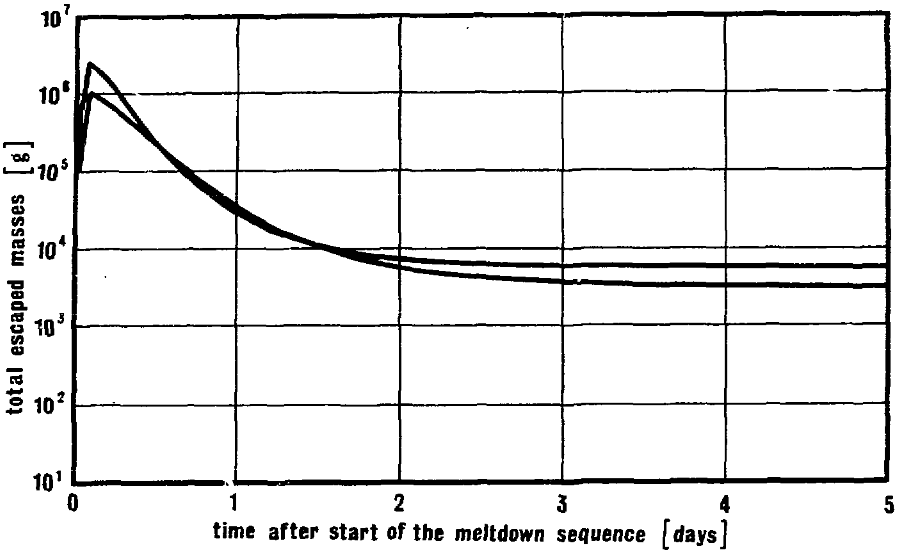

Fig. 6: Total escaped mass for 1000 kg and 2500 kg release cases

On the other hand a rough interpretation of Fig. 6 in its semi-logarithmic scale would state that the differences between both curves are negligible for risk assessment purposes. This surprisingly simple result, however, should by no means be generalized. Any change in the scenario might lead to unexpectedly differing results. The experience with aerosol behavior calculations tells that one has to use the code again when the input data change. No simple dependences of input and output exist.

One last problem remains to be treated. When an overpressure failure occurs the pressure in the containment relaxes from 9 Bar to 1 Bar. Consequently the sump water will boil vigorously until its temperature has decreased down to 100 °C. The question is how much of the aerosol mass that has been removed before from the containment atmosphere into the sump will be resuspended and will escape from the containment. A quick assessment of this process yields the following result.

The pressure relaxation causes 13% of the sump water to vaporize. During such a process the resuspended fraction of particles is by far not equal to the vaporized water fraction. So not 13% of the removed aerosol mass is resuspended but this value is divided by a retention factor.

$$\text{Resuspended aerosol fraction} = \frac{\text{vaporized water fraction}}{\text{retention factor}}$$

Very few measurements of such retention factors exist; they range from 10 to 10000. All these measurements have been done under conditions which are very different from a pressure relaxation boiling. For the moment a retention factor of roughly 1000 seems most likely. When a factor of 1000 is used the total escaped mass curves in Fig. 5 and 6 are not changed beyond the resolution of the graphic representation. A retention factor of 100 would increase the value of the escaped mass slightly. This leads to the preliminary assumption that the resuspension of particles from the sump during containment failure does not significantly contribute to the total aerosol mass escaped to the environment.

As a final check the values of the retention factor will be measured in the experimental part of the NAUA-program under realistic pressure relaxation conditions.

CONCLUSIONS

The NAUA code is presently the only one which calculates the behavior of particulate fission products in the containment of an LWR during core meltdown accidents. The code principally takes into account steam condensation phenomena which greatly enhance the removal of aerosols from the airborne state. The effects of steam condensation on aerosols have been measured and the results have been integrated into the code. However, the experimental determination of the amount of steam which is available for condensation onto the particles is not yet completed. The airborne mass and leaked mass curves given in this article should, therefore, be considered as upper limits

Calculations have been performed for a typical meltdown sequence with best estimate aerosol release functions. The airborne masses in the containment and the leaked masses have been evaluated. The leaked aerosol mass reaches its final value already after approximately 12 hours when the containment stays intact. When a containment failure is taken into consideration it would contribute significantly to the mass escaped from the containment during the first three days. So the containment integrity should be preserved at least for three days to keep the escaped amount of particulates as low as possible.

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ESTIMATES OF DOSE TO THE POPULATION WITHIN FIFTY MILES DUE TO NOBLE GAS RELEASES FROM THE THREE MILE ISLAND INCIDENT

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ABSTRACT

Estimates have been made of the dose to the population within 80 km (50 miles) due to noble gas releases from the Three Mile Island Unit 2 incident for the period March 28 to April 15, 1979. Source term, meteorological, and monitoring data used in these estimates were supplied by the Task Group on Health Physics and Dosimetry of the President's Commission on the Accident at Three Mile Island. The 22.5° sector-averaged form of the Gaussian plume atmospheric dispersion model was used to calculate doses due to immersion in air and inhalation. Our best estimate of the population dose to the total body is 15 person-sieverts (1500 person-rem).

INTRODUCTION

Beginning on March 28, 1979, a sequence of events occurred at the Three Mile Island (TMI) Unit 2 nuclear power reactor near Harrisburg, Pennsylvania, which resulted in the release of an amount of radioactive gases to the atmosphere in excess of that emitted during routine reactor operations. A comprehensive study of this incident has been prepared by the President's Commission on the Accident at Three Mile Island [1]. As part of this study the Task Group on Health Physics and Dosimetry requested the authors to estimate the dose to the population within 80 km (50 miles) of the reactor for the period March 28 thru April 15, 1979. Subsequent to these calculations, dosimetric monitoring data from around the plant were examined and adjustments were made in the population dose calculations. The purpose of this paper is to present our best estimate of the population dose from the TMI incident and to discuss the methodology used in making the calculation.

METHODS

AIRDOS-EPA

The AIRDOS-EPA computer code [2] was used to estimate the dose to the population within 80 km of the TMI plant. This code calculates downwind air concentrations using a constant mean wind velocity Gaussian plume atmospheric dispersion model [3]. The 22.5° sector-averaged form of this model as used in this study is given by

$$\chi = \frac{Q}{0.15871 \pi x \sigma_z \bar{u}} \exp [-1/2 (\frac{H}{\sigma_z})^2] \quad (1)$$

where

χ = ground-level air concentration (Bq/m^3) at downwind distance x (m),
Q = uniform radionuclide release rate (Bq/sec),
$\bar{u}$ = mean wind speed (m/sec),
σ_z = vertical dispersion coefficient (m), and
H = effective stack height (m).

The air concentrations calculated using Eq. (1) were used to estimate doses. The dose due to immersion in air is given by

$$D_{imm} = \chi C_{imm} (1 \times 10^{-6}) \quad (2)$$

where

χ = ground-level air concentration (Bq/m^3),
D_{imm} = air immersion dose (Sv),
C_{imm} = dose conversion factor for immersion in air (Sv/y per Bq/cm^3), and
1×10^{-6} = units conversion factor.

Doses due to inhalation were also calculated, but they were found to be insignificant when compared to the air immersion doses in this study. Doses were estimated for total body, red bone marrow, lungs, endosteal cells, stomach wall, lower large intestine wall, thyroid, liver, kidneys, testes, and ovaries.

The AIRDOS-EPA computer code also has the capability of estimating wet and dry deposition effects and the resulting doses from surface exposure and ingestion. Such calculations were not made for this study, however, since the only radionuclides considered were nonreactive noble gases.

Meteorological Data

Meteorological data taken at the TMI tower were obtained and adjusted for use as input in the AIRDOS-EPA calculations. Hourly values of wind direction, wind speed, and the vertical temperature gradient for the time period being considered were used. The temperature data were used to derive hourly values of the Pasquill atmospheric stability classes [4]. A joint frequency distribution of the average wind speed for each of 16 wind direction sectors and 7 stability classes was constructed.

Mixing height values for the period of the TMI release were not supplied by the Task Group staff. Instead, mean values of the mixing height for January and June were obtained [5] and averaged. The resulting value of 900 m was used in the AIRDOS-EPA calculations.

Source Term

Radionuclides from the TMI incident were emitted via a vent stack located atop the auxiliary building adjacent to the unit 2 containment building (Fig. 1). The stack is 55 m above ground level but only 5 m above the auxiliary building roof and 3 m above the closest obstruction. The stack is 1.2 m in diameter, and the effluent had an exit velocity of 36 m/sec. The temperature of the effluent was assumed to be near ambient [6].

A direct measurement of stack effluents during the TMI incident was not performed and thus the amount and identity of the aerosols released are unknown. It was assumed for dose calculational purposes that the release consisted of ^{88}Kr, ^{133}Xe, and ^{135}Xe. Other gases in the core inventory at the time of shutdown decayed rapidly during the first few hours, and made insignificant contributions to dose. The composition of the gas mixture as a function of time during the 19 day release was calculated using estimated quantities of the radionuclides in the core at shutdown and their half-lives [6]. It was found that of the total release 1% was ^{88}Kr, 95% was ^{133}Xe, and 4% was ^{135}Xe.

The total release of radionuclides used in these calculations was inferred from the response of a stationary gamma radiation monitor located external to the base of the stack. Release rate estimates (Bq/min) for various time periods during the incident were supplied to the authors by the Task Group staff. From this information, hourly release rates (Bq/hr) were generated assuming a linear change in the release rate between the data points supplied. The total release from this analysis was found to be 8.9 x 10^{16} Bq (2.4 x 10^{6} Ci). This total was distributed among the 16 wind direction sectors by assigning each estimated hourly release to the wind direction sector reported for that hour. The resulting release into each sector was apportioned among the three radionuclides considered as noted above and then dispersed out to a distance of 80 km using Eq. (1).

Population

The projected 1980 population within 80 km of TMI, adjusted for the actual 1979 population out to 3.2 km, was also supplied by the Task Group staff [6]. This area was divided into the 16 wind direction sectors and 10 annular distances: 0-1.6 km, 1.6-3.2 km, 3.2-4.8 km, 4.8-6.4 km, 6.4-8.0 km, 8-16 km, 16-32 km, 32-48 km, 48-64 km, and 64-80 km. In AIRDOS-EPA, the air concentration and subsequent individual dose is calculated at downwind distances at the center of each annular ring in each wind direction sector. This dose is then assumed to be received by each individual at that distance and direction. The resulting population dose for each sector and annular ring is the product of the individual dose and total population for that area.

RESULTS

Initial Calculations

Population dose estimates were prepared for the Task Group staff by assuming that the plume remained elevated during release. Plume rise due to the momentum of the emissions was taken into account. The total-body external dose conversion factors (Sv/y per Bq/cm^3) used in these estimates are 3.2 (^{88}Kr), 5.1 x 10^{-2} (^{133}Xe) and 3.8 x 10^{-1} (^{135}Xe)[7]. The resulting total body population dose by sector is shown in Table 1. While ^{88}Kr composed only 1% of the total release, its comparatively large dose conversion factor resulted in ^{88}Kr contributing up to 26% of the population dose in a given sector. The total population dose of approximately 4 person-sieverts (395 person-rem) is about a factor of 7 less than the 28 person-sieverts estimated from extrapolation of limited thermoluminescent dosimeter (TLD) measurements taken at the time of the incident [6].

Comparison of Observed and Predicted Doses

It is clear from Fig. 1 that the release point for the noble gases considered in this study is surrounded by buildings and other structures. As a result, downdrafts could at times have brought all or part of the TMI plume to ground level. Methods are available for estimating the effects of such downdrafts and building wakes on downwind air concentrations [8,9]. However, no such methods are available in AIRDOS-EPA.

Subsequent to the preparation of the population dose estimates for the Task Group staff, measured net dose values were obtained from twenty TLD's placed around the TMI site prior to the incident. These TLD's were located in various directions from the plant at distances ranging from 0.16 to 24 km. Comparisons were made between these measured values and values predicted using Eq. (1) assuming both an elevated release, as used above, and a ground-level (1 m) release. The latter release height was chosen to approximate the potential downdraft effects due to the presence of the buildings.

A summary of the results of these comparisons is shown in Table 2 [10]. It can be seen that the use of a ground-level release in the model results in a more favorable comparison than when the original elevated release condition is assumed in the model.

Revised Population Dose Estimates

Revised population dose estimates have now been made assuming a ground-level (1 m) release. The total-body dose estimates resulting from this calculation are also shown in Table 1. Revised estimates for other organs have been tabulated elsewhere [11]. The total dose to the population within 80 km is 15 person-sieverts, which is within a factor of two of that extrapolated from the TLD measurements (28 person-sieverts).

DISCUSSION

There is no universally accepted method for estimating health effects from radiation doses. The highest population dose estimated from the TMI incident (28 person-sieverts) is, however, only about one percent of the annual collective dose resulting from natural background (2400 person-sieverts). It has been estimated that the dose from TMI is too small to cause any detectable increase in cases of cancer, developmental abnormalities, or genetic ill-health [1].

There are a number of potential sources of error in these calculations that should be noted. The results of any Gaussian plume model calculation are directly proportional to the source term used as input if all other parameters are assumed constant. As a result, any error in the composition or magnitude of the assumed TMI source term will result in a like error in the dose.

The Gaussian plume dispersion parameters used in AIRDOS-EPA are based primarily on data measured over relatively flat terrain. As shown in Fig. 1, the TMI site is located in a river valley surrounded by rolling terrain. The Gaussian model may not perform as well under these conditions as it does for flat terrain [12].

More information is needed on the behavior of plumes around building complexes such as the TMI site. It is unlikely that the TMI plume was brought to ground 100% of the time during the release, but no information seems to be available on the behavior of the plume around the structures. Such information could help increase the accuracy of the dose calculations.

AIRDOS-EPA is designed primarily for estimating long term average doses from continuous releases of radionuclides, not relatively short-term releases like those considered here. The uncertainty associated with such short-term calculations is undoubtedly larger than the uncertainty associated with long term averages [12].

CONCLUSIONS

The AIRDOS-EPA computer code has been used to estimate the total-body dose to the population within 80 km due to noble gas releases from the TMI incident. These calculations are based on a 22.5° sector-averaged Gaussian plume atmospheric dispersion model assuming a ground-level release. The latter assumption was used because it resulted in better agreement between observed and predicted TLD doses than did use of an elevated release in the model. Our value of 15 person-sieverts is within a factor of two of the 28 person-sieverts estimated from extrapolation of TLD measurements without considering shielding effects due to dwellings. It has been estimated that the population dose received from the TMI incident is too small to cause any detectable physical health effects [1].

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Table I. Summary of estimated population doses to total body by sector resulting from the incident at Three Mile Island (March 28-April 15, 1979)

Compass direction[a]	Sector	Number of persons	Population dose (person-Sv) Elevated release	Population dose (person-Sv) Ground-level release
N	1	98,425	0.35	2.45
NNW	2	77,858	1.12	1.94
NW	3	162,267	0.63	2.83
WNW	4	106,277	0.53	1.54
W	5	96,229	0.22	0.64
WSW	6	50,221	0.03	0.16
SW	7	81,611	0.05	0.14
SSW	8	140,808	0.07	0.47
S	9	229,370	0.10	0.34
SSE	10	141,201	0.12	0.36
SE	11	70,570	0.04	0.10
ESE	12	233,336	0.06	0.16
E	13	173,341	0.06	0.28
ENE	14	250,668	0.10	0.46
NE	15	153,903	0.08	0.76
NNE	16	97,034	0.39	2.45
Total		2,163,119	3.95	15.08

[a] Wind "toward."

Table II. Summary of a comparison between predicted and observed doses resulting from the incident at Three Mile Island (March 28-April 15, 1979)

Height of release	Ratio $\left(\frac{\text{predicted dose}}{\text{observed dose}}\right)^a$ Range	Ratio $\left(\frac{\text{predicted dose}}{\text{observed dose}}\right)^a$ Median	Correlation coefficient, log (observed dose) vs log (predicted dose)[b]
1 m	5×10^{-2} -6.2×10^{0}	0.84	0.91
55 m	2×10^{-5} -2×10^{-1}	0.01	0.1

[a] A value of 1 signifies perfect agreement between predicted dose and observed dose.

[b] Maximum value = 1.

Fig. 1. Three Mile Island Unit 2 Nuclear Reactor Site Showing Vent Stack (Arrow) from which Noble Gases were Released During the Period March 28-April 15, 1979.

NEW DIRECTIONS FOR NRC'S REACTOR SAFETY RESEARCH PROGRAM

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ABSTRACT

The Energy Reorganization Act of 1974, which established the Nuclear Regulatory Commission, specified that NRC should carry out a program of confirmatory safety research that it deemed necessary to support its regulatory activities. Until recently, the bulk of the light water reactor safety research program was concentrated on confirming the adequacy of emergency core cooling systems to keep the fuel cooled during a large-break, loss-of-coolant accident. The Three Mile Island accident has focused attention on four major areas where more research is needed:

(1) better understanding of plant thermal-hydraulic behavior during anomalous transients and small-break LOCAs;
(2) more detailed studies of operator actions during accidents and more human engineering studies to improve the man/machine interface;
(3) better understanding of severely damaged fuel behavior; and
(4) better understanding of the integrity of plants during accidents that result in severely damaged or molten fuel

In response to these needs, a major part of the safety research program has been reoriented and new programs have been started. The thrust of these new directions is to elevate exploratory safety research to the same level as confirmatory safety research in NRC's program.

HISTORICAL PERSPECTIVE

The U.S. Government has sponsored safety research for nuclear plant safety assessment ever since the U.S. Congress established the Atomic Energy Commission (AEC) in 1946, and gave it the responsibility for promoting the development and utilization of atomic energy. The Division of Reactor Development was organized within the AEC on February 1, 1949, to manage the Commission's reactor development programs. The Reactor Safety Section of this Division was given the responsibility for establishing and managing a program of reactor safety research applicable to the reactor systems that were to be developed.

The Atomic Energy Act of 1954 further elaborated AEC's role in atomic energy development to include industry participation in the development of nuclear power for commercial applications, and the Act made the AEC responsible for regulating the use of atomic energy to protect the health and safety of the public.

The safety research tasks conducted as part of the early reactor experiments performed in the late 1940's became the base of an expanded safety research program in the AEC, with a broadening of the scope and test parameters. Early cooperation was initiated between RDD's Reactor Safety Section, the AEC's regulatory personnel in the Division of Civilian Application (later to become the Division of Licensing and Regulation) and the Reactor Safeguards Committee which was established as a statutory committee, the ACRS, in 1957. With input from these groups, the reactor safety research program was redirected to consider safety issues associated with the siting of central station power reactors outside the environs of government controlled sites and facilities.

By 1960, safety research programs had been established in the areas of reactor kinetics, hazardous and energetic chemical reactions, fission product release, and reactor containment covering a range of test parameters with different reactor fuels and containment and reactor models to encompass both fast and thermal reactors.

The SPERT series of reactors, located at the Idaho National Engineering Laboratory, performed overpower and excursion tests (1954-1971), including tests which involved core destruction and excursion tests from full power PWR operating conditions. The purpose of the SPERT tests was to study the safe limits of uncontrolled nuclear excursions, including an understanding of various self-shutdown mechanisms (Doppler effect, void formation, etc.), that prevented potential nuclear excursions from reaching dangerous levels. Over 3,000 tests were performed with the four SPERT reactors providing data for the development and verification of analysis methods which now permit the prediction of nuclear excursion behavior of light water reactors with a high degree of confidence in the expected results.

Other safety research work initiated during that period included chemical reaction studies, fission product release experiments, containment testing, analysis of the effects of earthquakes on structures, and fuel tests in the TREAT reactor.

During the decade of the 1960's, the reactor safety research program underwent significant changes as utilities began to respond to economic incentives offered by the AEC and the reactor suppliers by ordering commercial power plants with power ratings up to 1,000 MWe and attempting to site the plants closer to highly populated load centers. A factor that gave impetus and renewed interest in reactor safety research was the occurrence of the SL-1 accident in January 1961 which resulted in the death of three reactor operators. With increased attention focused on reactor safety, the Commission reorganized the safety research functions in the Division of Reactor Development by creating the Office of Nuclear Safety in August 1961, and increasing the safety research staff. The importance of directing attention to emergency core cooling studies was emphasized in various safety reviews of large reactor systems including the Brookhaven study of hypothetical accidents in large reactor systems to update Price-Anderson accident indemnity levels (1965-1966) and the Ergen Task Force study of Emergency Core Cooling (1966-1967).

In 1962, the AEC initiated a test program at the INEL to study the consequences of a loss-of-coolant accident (LOCA) in a commercial light water reactor caused by a major primary pipe rupture. This was primarily intended to be a study of the release, transport and deposition of fission products released from a reactor core meltdown and was known as the LOFT-U program. The plan, never undertaken, was limited to a few nonnuclear blowdowns (simulated pipe rupture), and a reactor core meltdown. The facility included a pressurized water reactor (PWR) experimental reactor test facility.

In the mid-1960's, the size of commercial nuclear power plants rapidly increased to about 1,000 MWe. As a result, additional engineered safety features had to be developed for large reactor cores in order to prevent releases of radioactivity to the environment should a LOCA and a consequent reactor core meltdown occur. In 1967, large LWR plant designs began to include new safety features, such as Emergency Core Cooling Systems (ECCS) for core flooding and to prevent reactor core meltdown in the event of a LOCA. At that time, the AEC redirected the LOFT program toward the complexities of studying the new safety design features for large reactor systems and cores, rather than a reactor core meltdown. This resulted in the need for an integrated test facility to study the nuclear, thermal, hydraulic, and structural processes associated with a LOCA in a PWR. Consequently, in 1969, a complete redesign of LOFT was begun so that the system would model as nearly as possible the conditions present in a primary coolant system and core of a typical large PWR and associated ECCS, as well as providing systems to contain the possible release of radioactive fission products. Aside from the normal PWR systems, LOFT has a special tank to contain the blowdown and process the release of fission products from the experiments. Since LOFT was essentially completely designed and partially completed in accordance with the earlier objectives, significant design changes and almost a new start were required to satisfy the new requirements.

Other large scale safety experiments that were started in the 1960's included the Nuclear Safety Pilot Plant (1962) to study fission product aerosols and their control by chemical sprays and charcoal filter systems; the Containment Systems Experiments (1963) for depressurization testing (blowdown) and fission product transport and control experiments; and the Power Burst Facility (1964) for reactor transient testing of fuel assemblies up to meltdown. Additional engineering scale tests started in the 1960's, included the FLECHT fuel model subassembly tests for heat transfer and reflood tests following a loss of coolant, the Semiscale tests for modeling LOFT blowdown and ECC and checking analytical code capabilities, pipe rupture studies to determine limits on crack initiation and propagation, and the heavy section steel technology program.

The decade of the 1970's began with nuclear power, nurtured under the umbrella of the Atomic Energy Commission, appearing more and more attractive economically. As a result of oil shortages and problems with burning coal, a number of utilities ordered nuclear plants without fully understanding their safety problems, and probably believing that the government under the AEC was going to continue to nurture the industry. At the same time, public interest groups, spawned during the turbulent years of the Vietnam War, began to challenge the assurances given by the AEC and the nuclear industry that reactors were safe. In particular, these challenges revealed that the industry and the AEC had not done their homework adequately regarding the margin of conservatism in ECCS design and operability in large nuclear plants. In order to provide a basis for continued licensing activities, the licensing staff published the "Interim Acceptance Criteria" for ECCS in June 1971. The Interim Criteria imposed a set of guidelines for the safety analysis and operation of ECCS which was intended to limit the consequences of a loss-of-coolant accident, should one occur. In January 1972, the AEC initiated a public rulemaking hearing on the Interim Acceptance Criteria to permit an airing of public and regulatory staff viewpoints and differences of opinion regarding their acceptability for licensing nuclear plants.

The hearings revealed shortcomings in the adequacy of safety research data and the ability of computer codes to fully describe and quantify the course of events during a loss-of-coolant accident. However, the revised Acceptance Criteria for Emergency Core Cooling Systems were generally found acceptable for the conservative safety analysis and licensing approval of light water nuclear plants, and with some minor modifications they were published as part of 10 CFR Part 50 on December 28, 1973.

After reviewing the hearing testimony and taking into account various criticisms of its programs, the Commission reorganized the AEC safety research programs by establishing in May 1973 a new Division of Reactor Safety Research which was intended to be more responsive to the needs of the licensing staff and to be free of the appearance of competing for funds with the growing fast breeder development program.

The new Division of Reactor Safety Research increased the research work on loss-of-coolant accidents by directing its program to the safety questions that led to the publication of the ECCS Acceptance Criteria. Code development work was expanded to improve existing models for regulatory use as "evaluation models" and "best estimate" models for use by research personnel for code assessment through the analysis of experiments. All aspects of a loss-of-coolant accident were carefully evaluated and safety programs initiated to provide the greatest payoff toward the assurance that ECCS would work. New programs were started on ECC downcomer mixing and steam bypass, on heat transfer and coolant phenomena during depressurization and reflood using full scale electrically heated models of fuel assemblies, on the thermal shock effects on primary vessels during depressurization and early ECC injection, and on fuel damage during a loss-of-coolant accident to determine the extent of fuel channel blockage, and to study metal-water reactions and close embrittlement. The Semiscale experiment was modified to more closely represent and model LOFT and, by comparing the data between the two systems, to permit a more valid extrapolation to the characterization and prediction of accident behavior in full scale reactor systems.

In early 1972, at the prodding of the Joint Committee on Atomic Energy who wanted a basis for renewal of the Price-Anderson Act, the Commission contracted with Professor Norman Rasmussen of MIT to direct a study of the risks from accidents at nuclear power plants. This study proved to be far more difficult than originally envisaged, but under the intellectual leadership of Rasmussen and Saul Levine, AEC's staff director, the methodology was developed to identify the dominant accident sequences leading to release of radioactivity from nuclear plants and to quantify the risks from such accidents. The final report, WASH-1400, was published in late 1975, and it immediately became a source of controversy. Proponents of nuclear power used the report to reassure the public of the safety of nuclear power, while opponents attacked the methodology and the probability estimates as being too unreliable for meaningful risk estimates. A subsequent revew in 1978, by the Risk Assessment Review Group, chaired by Professor Hal Lewis, found that the risk assessment methodology was sound and should be used more widely in the regulatory process but that, due to an inadequate data base, the error bounds on the risks quoted in WASH-1400 were greatly understated.

Seemingly lost in the controversy surrounding WASH-1400 was the fact that the pioneering methodology developed was a giant step in establishing a framework for making reactor safety evaluations more rational. Perhaps more surprising was the fact that the lessons from WASH-1400, that transients, small LOCAs and human errors are important contributors to overall risk, were not adequately reflected in the safety research program.

THE ROLE OF RESEARCH IN NRC

Closer ties between the AEC's safety research programs and the needs of its regulatory divisions were established on October 11, 1974 when President Ford signed into law the Energy Reorganization Act of 1974 which provided for the abolishment of the AEC and the assumption of its regulatory and safety research functions by the Nuclear Regulatory Commission. The 1974 Act provided for an Office of Nuclear Regulatory Research within NRC, and AEC's Division of Reactor Safety Research was integrated into this Office. The Office of Nuclear Regulatory Research was authorized to perform research characterized as "confirmatory assessment" which was intended to relate specifically to regulatory decisions for the safe and environmentally compatible operation and protection of nuclear facilities and materials - as distinguished from the research and development functions assigned to ERDA which included the operating responsibility for NRC's safety research facilities.

Thus, at the beginning of 1975, the newly formed Nuclear Regulatory Commission found itself in the somewhat anomalous position of being an independent regulatory agency with a research program whose size, in terms of budget, was as large as the rest of the agency combined. It is not surprising that some in NRC were not totally at ease with a large research program whose role in the agency was not completely clear. Over the past 5 years, it has become accepted that the research program provides essential support for NRC's regulatory activities by providing a technical basis for the licensing decisions, regulatory guides and standards that are the main responsibility of the agency. Many of the regulatory judgments that must be made by NRC involve the assessment of potential accidents far outside the range of normal engineering experience, and this in turn requires a thorough understanding of the accident phenomena which can only be provided by the safety research program. Without this underpinning of basic safety information and understanding, NRC's regulatory judgments could be increasingly challenged in our hearing processes and perhaps in the courts as well.

SOME LESSONS FROM THE TMI-2 ACCIDENT

Under the NRC the LWR safety research program has doubled in size, growing from $62 million in FY 1976 to $125 million in FY 1980. Before the TMI accident, the bulk of this work was aimed at answering the backlog of questions on loss-of-coolant accidents that arose during the ECCS rulemaking hearing. Among the staff in NRC it was an article of faith that all of those questions had to be answered. In a sense it was viewed as a mortgage that had to be paid off in order to allow reactors to keep operating. The safety research program was intensely focused on measuring the effectiveness of emergency core cooling systems in the event of a large-break loss-of-coolant accident, and the program has been quite successful along these lines. The culmination of years of planning and testing has confirmed the margins of conservatism in ECC systems if they function as designed during an accident.

But one unfortunate result of the intense focus on large break LOCAs was that the research program paid very little attention to accidents like operational transients and small leak loss-of-coolant accidents which WASH-1400 showed were both more likely and contributed more to the overall risk than large LOCAs. One of the primary lessons of the TMI-2 accident is that the safety research program was not properly balanced. Prior to the TMI accident, the research program was in lock-step with the regulatory approach to licensing nuclear plants, which was to define design basis accidents thought to be severe enough to blanket all lesser accidents. That is, if plants were designed to accommodate design basis accidents, it was thought that all other accidents believed to be even remotely likely could also be accommodated by the plants' design features. Where the TMI accident revealed a weakness in the regulatory approach, and thereby in the safety research program as well, was in the fact that there was a whole class of accidents that NRC had not considered adequately--accidents that begin as normal operational transients but due to actions by the operator, where he is misled by his instruments or otherwise misunderstands what is happening, the automatic safety systems are bypassed or overriden. These accidents can lead to severely damaged cores and, if the operator does not take the right corrective actions, even to core meltdown and widespread release of radioactivity to the environment. Thus, a major impact of the TMI accident, if not the most significant, has been to dispel the complacency of the NRC and the nuclear industry that had built up over the years. Suddenly, the regulatory process has to deal with the fact that core meltdown accidents are not hypothetical and that emergency planning and evacuations are not remote contingencies.

NEW DIRECTIONS FOR SAFETY RESEARCH

Within a few months after the TMI accident, a major part of the ongoing safety research program had been reoriented to investigate some of the questions raised by the accident. The most obvious change was to reorient the major thermal-hydraulic test facilities away from large break LOCAs toward the study of operational transients and small break LOCAs. For example, the mission of the LOFT program was changed from the study of ECCS effectiveness during large LOCAs to the study of overall reactor system behavior, including operator actions, during operational transients and small break LOCAs.

In addition to the ongoing programs, it is clear that major new areas of safety research must be undertaken. NRC intends to carry out research in each of the areas outlined below, and in some instances programs have already been started.

I. Better Understanding of Operational Transients and Small LOCAs
  - Separate effects and thermal-hydraulic tests
  - Integral tests in LOFT
  - Improved, fast-running computer codes
  - Establish a data base for each operating reactor

II. Enhanced Operator Capability
  - Improved plant instrumentation
  - Improved control room display and diagnostic systems
  - Improved operator training simulators

III. Plant Response Under Accident Conditions
  - Coolability of severely damaged cores
  - Release and transport of fission products
  - Better understanding of coolant chemistry after accidents
  - Hydrogen behavior in coolant and containment
  - Effect of hydrogen explosions on structures
  - Maintaining containment integrity under fuel melt conditions

IV. Post-Mortem Examination and Plant Recovery
  - Examine TMI damaged fuel
  - Measure fission product chemistry and plateout
  - Examine TMI safety related electrical equipment

V. Improved Risk Assessment
  - Identify dominant accident sequences for each operating plant
  - Assess site specific accident consequences
  - Analyze human error rates
  - Analyze operational failure data

VI. Improved Reactor Safety Features
  - Study improved containment concepts
  - Study improved decay heat removal systems

CONCLUSION

The new research programs outlined above will, when completed, provide a vastly improved understanding of plant behavior during accident conditions and will provide plant operators with better information to cope with the accidents. But the research cannot, by itself, improve the safety of reactors. It can only provide the understanding and information to be used by the designers, operators and regulators in their responsibilities for assuring safe nuclear power.

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INSTITUTE OF NUCLEAR POWER OPERATIONS (INPO)

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ABSTRACT

The electric utility industry established the Institute of Nuclear Power Operations, or INPO, the purpose of which is to ensure the highest quality of operations in nuclear power plants. INPO will be an industry self-help instrument focusing on human factors. From top management to the operator trainee, it will measure utility performance against benchmarks of excellence and help utilities reach those benchmarks throughout training and operating programs. INPO will see that the utilities ferret out lessons for all from the abnormal operating experiences of any. It will do everything possible to assist utilities in meeting its certification requirements, but will have the clout to see that those requirements are met. INPO is also managing the nationwide system of utility emergency response capability.

INTRODUCTION

The Institute of Nuclear Power Operations (INPO) is a not-for-profit organization sponsored by the nation's nuclear utilities to ensure excellence in operation of nuclear power plants. INPO opened its doors in Atlanta on December 3, 1979, and currently has about 50 staff members on-board, with a projection of about 200 by the end of 1980.

FORMATION

The accident at Three Mile Island made obvious to industry leaders the importance of plant operating safety and the enormous cost of mistakes. The industry response to TMI included the coordinated efforts of individual utilities, the Electric Power Research Institute, the Atomic Industrial Forum, the Edison Electric Institute, the American Public Power Association, the National Rural Electric Cooperative Association, and the suppliers. Leadership for their activities was provided by the TMI Ad Hoc Nuclear Oversight Committee, chaired by Floyd Lewis of Middle South Utilities. In addition to direct response to the accident, the oversight committee guided the formation of three new organizations: EPRI's Nuclear Safety Analysis Center, the Nuclear Electricity Insurance Limited, and the Institute of Nuclear Power Operations.

Subsequent progress has been steady. On June 28, 1979, Floyd Lewis announced that detailed plans for the Institute of Nuclear Power Operations would be developed under the leadership of Dr. Chauncey Starr, Vice Chairman of the Electric Power Research Institute. In late August and early September, a series of regional meetings was held to present the INPO concept and plans to the nation's utility executives. They expressed strong support for the formation of INPO, and INPO was incorporated in Delaware on October 12, 1979. The new INPO Board of Directors, chaired by William S. Lee of Duke Power Company, selected Atlanta as the site for INPO and Dennis Wilkinson, Vice President of Data Design Laboratories and first commanding officer of the USS Nautilus, as the president of INPO. The move into permanent headquarters, only a few hundred yards from the temporary location, is scheduled for early May.

STRUCTURE

The INPO organization has a conventional structure of a Board of Directors, Officers, Four Technical Divisions, and an Administrative Division. In addition, an Advisory Council, made up of nationally-prominent educators, scientists, engineers, and health specialists, will provide independent input into INPO's efforts. Also, an Industry Review Structure will assist in reviews of the products generated by our technical divisions.

The eleven-member Board of Directors is composed of utility executives. Six members are representatives of investor-owned utilities, one member represents TVA, a federal government utility, one member represents non-federal public utilities, one member represents cooperative utilities, and two members are specifically selected as persons with current, or recent, plant operating experience.

Members of the BOARD are:

William S. Lee, Chairman of the INPO Board of Directors
President, Duke Power Company

G. Carl Andognini, Superintendent of Nuclear Operations
Boston Edison Company

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The ADVISORY COUNCIL is composed of 17 members who are recognized leaders in their fields. They are:

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Dr. Robert Seamens: Dean of Engineering, MIT; also the Henry R. Luce Professor of Environment and Public Safety.

Dr. John A. Swartout: Retired Vice President, Union Carbide Corporation; Former Deputy Director of Oak Ridge National Laboratory. Currently, Chairman of Utilities Scientific Advisory Council to Nuclear Safety Analysis Center.

Dr. M. Gordon Wolman: Chairman, Department of Geography and Environmental Engineering at Johns Hopkins University.

INPO has Four Technical Divisions:

The TRAINING AND EDUCATION DIVISION's activities will include program recommendations for developing operating personnel and the technical staff. Significant efforts will be directed toward management and supervisory training with special emphasis on improving operational safety.

Curricula, lesson plans, and training materials will be reviewed and existing approaches will be upgraded. Instruction programs will be accredited and assistance will be provided for the training of instructors and the development of teaching skills. Workshops and seminars will be conducted to assist in the development of management and supervisory personnel and the instructional staff.

Initial emphasis is being placed on identifying and making available the best of existing training practices, materials, and resources. Also, specific operating experiences, like those of TMI, will be fed back into the training programs.

The CRITERIA AND ANALYSIS DIVISION is developing benchmarks of excellence which will be the basic tools of INPO. Information collected from present utility practices is being used as the basis for these benchmarks. The benchmarks will define optimum performance rather than minimum acceptable standards. A dynamic approach is being applied for the development of the benchmarks, with continual refinement expected through experience and knowledge gained during INPO evaluations.

As an integral part of benchmark developmen , operating experiences throughout the industry will be reviewed and used to help perfect the benchmarks: Licensee Event Reports (LERs) and other operating reports will be analyzed by INPO. Causes, effects, and appropriate remedial and preventive actions will be considered. Operating experiences, and those superior methods, practices, and procedures observed during INPO evaluation efforts will be shared across the industry.

Additionally, the Division will sponsor studies in direct support of operations and provide liaison with architect-engineers and vendor organizations.

The most important task of INPO is to perform periodic, detailed evaluations of every member's nuclear power plant operations. The EVALUATION AND ASSISTANCE DIVISION has this specific charge. Every plant operation will be evaluated against the benchmarks of excellence, coupled with common sense and the experience of evaluators who are aware of on-line problems of nuclear plant operations.

The first pilot evaluation was performed during February, the second pilot evaluation was performed in March, and the final pilot evaluation is being performed now; routine evaluations will begin during May. Two teams are presently being prepared for the evaluations. It is important to note that these teams are composed of experienced operating professionals. Each member has had years of actual power plant operating experience. INPO is striving for six such teams to be in operation by the end of the year. Presently, these teams include many persons on loan from member utilities and probably will always entail some loaned personnel. The use of loaned persons should permit, and promote, rapid exchange of safe operational practices throughout the industry. By having loaned persons for one or two years on the teams, they will have the opportunity to look at many plants, to see the good and bad aspects. This knowledge will be taken home as a good investment by the parent utility that loaned them to INPO.

Evaluations of each plant will be performed on approximately an annual basis. The plan is to complete at least one evaluation of every operating nuclear power plant by July, 1981.

The industry was not prepared in advance to respond to the Three Mile Island accident a year ago, and a great effort has been made on many fronts to become prepared for the future. One INPO Division is devoted to EMERGENCY PREPAREDNESS. This Division will perform specific functions related to response to any accident that might occur.

A model emergency response plan for utilities was developed last summer under the leadership of an AIF subcommittee. This plan was transmitted to all utilities, and a workshop was held last fall on its use. Continuing review to keep the model plan up to date and evaluate its appropriateness will be performed by the EMERGENCY PREPAREDNESS DIVISION. A manpower listing of experts in various facets of plant operation, and how to acquire their help, will be maintained and made available to assist any utility for an emergency. The list will include a description of individual qualifications and abilities to assist in quickly matching the proper person with the immediate problem. This Division will also maintain an inventory list of emergency equipment, where it is located, and whom to contact concerning availability.

These technical divisions have the support of an ADMINISTRATIVE DIVISION to provide financial, personnel, communications, and general office management functions.

CURRENT STATUS

Expressions of support for INPO's operations have come from many quarters. The President established a special Commission, chaired by Dr. Kemeny, to investigate the Three Mile Island accident. The Report of the President's Commission on the Accident at Three Mile Island, better known as the Kemeny Commission Report, made reference to INPO in its recommendations. It stated that INPO may be an appropriate vehicle for establishing and implementing a program for the utility industry to take a stronger role to ensure the effective management and safe operation of nuclear power plants.

A separate investigation was performed for the NRC, and their Special Inquiry Group report, better known as the Rogovin Report, stated that, "We believe INPO can play an especially important role in providing affirmative assistance to its members to upgrade the competence of their site crews and management assistance the NRC cannot easily provide without compromising its enforcement role."

Especially important is the broad support for INPO from the total nuclear utility industry. Public utilities, cooperative utilities, and investor-owned utilities are all supporting INPO. Already 55 utilities are members of INPO. Tennessee Valley Authority, a federal utility, is a member. Sacramento Municipal Utility District (SMUD), and Salt River Project, non-federal public utilities, are members. Dairyland Power Cooperative is a member, and most investor-owned utilities are members. There are only seven more to hear from; no one has refused to join. The united goal of safety in nuclear power plant operation is supported by all.

The current period is an important one for INPO. Recruiting is a top priority; leading professional staff members are now being chosen. Within the next few weeks, routine operation of the annual evaluation teams and of the operating experience analysis program will start. The effect of INPO on operating safety will be a cumulative process, and the strong industry support has provided a solid foundation for INPO's activities.

THE PRESENT STATUS OF NUCLEAR WASTE MANAGEMENT IN THE FEDERAL REPUBLIC OF GERMANY

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ABSTRACT

The German Government and federal authorities have stipulated in a special regulation that new nuclear power stations will only be licensed if safe waste management ("Entsorgung") is guaranteed. Hence the energy-producing utilities commenced the planning of a national fuel-cycle-centre ("Entsorgungszentrum") which includes installations for reprocessing and ultimate storage. Detailed studies have shown that from a safety engineering viewpoint, there is no objection to this waste management concept. However, for political reasons, its realization is not acceptable at present. Hence the Federal Government considered alternative solutions, e.g. the contruction of away-from-reactor, temporary storage facilities and a small scale reprocessing plant.

INTRODUCTION

Secure energy supplies are essential for the continued existence and development of an industrial society such as the Federal Republic of Germany. At present, almost 60 % of our country's primary energy requirement is met from imports. It is a well-known fact that fossil fuels, such as coal, oil and natural gas, cannot meet our energy needs in the long term. Moreover, the exploitation of regenerative energy sources, such as the sun and wind, or the production of hydrogen as a substitute for petrol, is still only at the research and development stage. In contrast, nuclear energy is an available source of power which can be produced economically and on an industrial scale. Within the framwork of extending its energy programme, the Federal Government feels that further limited development of nuclear energy is essential in addition to giving priority to the use of coal. However, this depends on the establishment of a safe waste management system for nuclear power stations.

WASTE MANAGEMENT PROVISIONS

The German term "Entsorgung" means "waste management" and covers all the fuel processes following the reactor stage. On the basis of principles established jointly in 1977 by the Federal and State governments [1] , waste management involves:

- the safe and correct storage in a suitable place of all the irradiated fuel elements resulting from the operation of nuclear plants,

- the utilisation of these fuel elements through reprocessing, or in special cases, their treatment for final storage without reprocessing,
- the treatment and removal of any resultant radioactive waste.

This definition of waste management carries the force of law. Consequently, approval will only be given for the construction and operation of nuclear power stations if the operator can demonstrate adequate waste management provisions in accordance with these principles. In the first instance, this means binding agreements on the safe storage of the irradiated fuel elements for a period of at least six years, with extensions to cover the whole of the plant's operating period. However, until this waste management concept is implemented (including the processing and final storage of radioactive fission products) it is sufficient to be able to demonstrate the existence of intermediate storage facilities for fairly long periods at home and abroad.

This waste management strategy was confirmed by the resolution of 29 February 1980 made by the heads of the Federal and State governments, although certain new points were added: thus, the re-utilisation of irradiated fuel elements after reprocessing and direct conditioning for final storage are regarded as equally important. From 1 January 1986, there will be an additional requirement for the issue of provisional partial construction permits for new nuclear power stations in that progress will have to have been made in the establishment of at least one reprocessing plant or of a plant for the treatment of spent fuel elements for final storage without reprocessing. In addition, the planning procedure for the projected final store in Gorleben (Lower Saxony) must be continued and there must have been progress in the examination and preparation of the store.

Special waste management rulings will be drawn up for the prototype reactors, the high temperature reactor THTR-300 and the fast breeder reactor SNR-300 which are currently under construction.

NUCLEAR WASTE MANAGEMENT IN THE FRG

The aim nuclear waste management. In order to close the fuel cycle, the Federal Goverment devised a waste management concept in 1974, to be operated jointly by the Government and by industry. It covered all stages of fuel element storage, from reprocessing and fuel recycling to waste treatment and final storage. This waste management concept for light water reactors takes as its principal aim the reprocessing of spent fuel elements, in order

1) to convert for ecological reasons the resultant radioactive fission products into a form which will guarantee their safe final storage over the necessary periods and thus their exclusion from the bio-cycle
2) to recover and recycle the fissionable fuel (uranium, plutonium) from spent fuels element.

I.e. if plutonium and uranium from a 1400 t/y reprocessing plant are recycled in thermal reactors an amount of 40 million t hard coal units per year can be saved (2,4 billion t/y when the fissionable fuel is recycled in fast reactors).

The accumulation of irradiated fuel elements. The Federal Republic of Germany currently operates 10 commercial light water reactor power stations with an installed electrical output of 9 GW. It can reasonably be assumed that this generating capacity will increase to almost 30 GW_e by 1990 and to over 50 GW_e by the year 2000. This in turn means that the annual production of irradiated fuel elements will increase from the current level of 230 tonnes of uranium to about 750 tonnes by 1990. Over the same period, the quantity of spent fuel to be stored will have accumulated to over 6000 tonnes of uranium.

As has already been mentioned, the only facilities in the FRG for reprocessing irradiated fuel elements from light water reactors is the reprocessing plant in Karlsruhe (WAK). However, with its installed annual throughput of 35 tonnes of uranium, it can make only small contribution. The contracts with the French company COGEMA (Compagnie Géneral des Matières Nucléaires) in Cap la Hague, which will have taken a total of 2300 tonnes by the year 1989 for subsequent reprocessing, make a considerably greater contribution towards solving the waste management problem. This will clearly relieve the situation, at least in the short term.

The capacity for the storage of spent fuel elements in nuclear power stations will be increased at the beginning of the 80's: this is attributable to the installation of so-called <u>compact storage frames</u> in the decay basins, which will increase the capacity of these basins by many times from the present level of about two discharge batches. Safety problems associated with the increase in activity levels or the production of decay heat are only of subsidiary importance, since both factors are determined primarily by the last batch to have been discharged.

<u>The waste management concept.</u> The steep increase in available storage capacity as a result of the introduction of compact storage has considerably relieved the waste management situation in the short and medium terms. This form of storage was first introduced in the FRG at the beginning of this year when the compact storage facilities in the Biblis A and B nuclear power stations were lincensed. There are also plans for conversion of existing nuclear power stations in the near future. However, it will not be possible to introduce compact storage in two nuclear power stations and alternative plans have been made for the provision of the required storage capacity. From now on, all future nuclear power stations will be equipped with compact storage facilities. Excluding plants operating on an experimental basis, the WAK is at present the only plant of its type in the FRG. This demonstration plant is of particular importance in the planning of a large-scale German plant. Since 1971, it has been used successfully to reprocess fuel elements from various experimental and production reactors with burn-up of up to 39000 MWd/tU.

A large proportion of the irradiated fuel elements produced in the Federal Republic of Germany is reprocesses or initially put into interim storage in the <u>French reprocessing plant at Cap la Hague.</u> In addition, the contracts with the French also provide for conditioning of the resultant highly radioactive waste by vitrification and its subsequent return to the Federal Republic of Germany. Irrespective of the decision on a site for the National Nuclear Fuel Cycle Centre, it is planned to use the final storage facilities in the salt dome at Gorleben to deposit this waste. The same applies to radioactive waste from the Karlsruhe reprocessing plant.

The current waste management strategy is thus based mainly on interim storage of the spent fuel elements with subsequent reprocessing abroad. In the medium and long term the National Nuclear Fuel Cycle Centre alone should be able to cope with all the fuel elements from German nuclear power stations. The planned plant, which will have an annual throughput of 1400 tonnes of uranium will encompass all areas of waste management, from fuel element storage, reprocessing, uranium- and plutonium-processing and final conditioning to final storage of the radioactive waste materials in a salt dome on the site.

At the same time, alternative solutions are being examined. In contrast to the integrated concept, these alternatives envisage, for the present, two regional interim storage sites, in North Rhine-Westphalia and Lower Saxony, each with a capacity of 1500 tonnes of uranium. Moreover, the State of Hesse has said that it is prepared to provide a site for a reprocessing plant with an annual capacity of 350 tonnes. At the same time, the direct final storage of spent fuel elements without reprocessing is being examined from the point of view of feasibility and safety. However, since as yet we have only the initial results of this concept from Sweden [2] at our disposal, a final decision cannot be expected before the mid-80's.

Responsibilities Whilst it is the electricity supply companies who are responsible for all stages of the operation from storage of the fuel elements to waste treatment, the Federal Government is responsible, under the terms of the Atomic Energy Law, for final storage. The costs of the individual stages, including final storage, are allocated according to responsiblility, i.e. according to the actual user. In order to implement the waste management concept, twelve electricity supply companies who operate nuclear power stations, formed the Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen mbH (DWK). The Federal Government commissioned the Physikalisch-Technische Bundesanstalt, Braunschweig (PTB) with the setting up and operation of the final waste store.

Licensing procedures The Atomic Energy Law forms the legal basis for the licensing of nuclear plants. The actual State in which the plant is to be sited is responsible for the licensing procedure. The Federal Minister of the Interior (BMI), who is responsible for atomic safety and protection against radiation, supervises the approval activities of the State authorities, and also has the power to issue recommendations. The BMI is advised by two independent committees of experts, the Reactor Safety Commission (RSK) and the Radiation Protection Commission (SSK), whilst the State authorities also have independent experts, such as the Technical Control Board (TÜV), the Association for Reactor Safety (GRS), to assess applications from a safety point of view.

THE INDIVIDUAL STAGES OF THE WASTE MANAGEMENT PROCESS

National Fuel Cycle Centre An integrated Fuel Cycle Centre, at which all the main stages of the fuel cycle are handled in one place, represents the optimum solution for the forseeable future both in terms of safety (e.g. minimisation of transportation, greater protection against terrorist mis-use) and economic factors. This viewpoint has also been confirmend by the International Nuclear Fuel Cycle Evaluation report (INFCE), among others. To realise this type of complete waste management system, the existence of a suitable salt dome is absolutely essential. Since there are a large number of potentially suitable salt domes in Northern Germany (Lower Saxony) which will, in all probability, provide the necessary conditions for final storage, Germany is not considering the use of other geological formations, such as basalt, granite or clay, for the final storage of highly radioactive waste, although their suitability is being investigated in other countries. Concerning the disposal of low- and medium-active waste, experiences from the ASSE salt mine are available. Here since 1967 different techniques have been tried out successfully. At the beginning of 1977, the Government of Lower Saxony put forward the salt dome at Gorleben as a possible site for the final store and thus for the National Fuel Cycle Centre. Applications for the construction and operation of the Fuel Cycle Centre followed shortly afterwards. These applications are based on a plant with a total throughput of 1400 tonnes of uranium per year [3]. It will handle the following individual stages:

- Fuel element storage: the applications provide for the construction of 6 wet storage basins, each with a capacity of 500 tonnes.
- Reprocessing plant using the Purex process: the four-line plant will have an average annual throughput of 4 x 350 tonnes; peak output is 4 x 2 tonnes of uranium per day. Average burn-up is 35000 MWd/day.
- Further processing of the recovered atomic fuels: all the plutonium recovered in the centre will be used to produce mixed-oxide fuel elements.
- Conditioning of the resultant radioactive waste by cementation. For solidifying highly active fluid waste, it is planned to use the AVM-vitrification process (AVM = Atelier de Vitrification de Marcoule), which has been tested in France and found to be successful. The plant will produce 140 m^3 of glass annually, which is equivalent to a daily output of between 4 and 5 x 70 1 glass blocks. The PAMELA process, which was developed in the FRG, is being considered as an alternative; in this case, the glass, in the form of minute beads, is sealed in a lead matrix.
- The last stage is the final storage of all resultant radioactive waste products. This amounts to a total of about 60.000 m^3/year, with low- and medium-active waste accounting for the greater part of this volume.

In October 1977, after detailed examination of the safety report submitted by the applicants and of numerous other documents, the RSK and the SSK published a joint statement on the feasibility of the National Fuel Cycle Centre from a safety point of view [1]. Among other things, it indicated that, on the basis of present knowledge and in terms of the suitability of the site and of the plant to be constructed for fuel element storage, there could be no reservations as to the safety of the scheme for the reprocessing and final conditioning of radioactive waste and for the processing of uranium and plutonium.

The concept of the final storage of radioactive waste in salt domes is a good solution in terms of safety, since it permits the permanent and reliable exclusion of the waste from the biosphere. The extent of the salt dome at Gorleben (approx. 80 km^3) will allow the storage of low- and medium-activity waste. Moreover, it is also expected that there will be sufficiently large quantities of rock salt to cope with the heat-producing, highly active waste. However, final confirmation of this will not be possible until exploratory drillings as well as shafts and workings, have been made.

In connection with these findings, the Government of Lower Saxony organised a one week Gorleben symposium at the beginning of 1979. It was attended by over 60 German and foreign experts representing both supporters and opponents of the waste management concept. Subsequently, in May 1979, it concluded that, from a safety point of view, the integral waste management concept was feasible, but pointed out that it could not be realised for the present for political reasons. Nevertheless, the Government of Lower Saxony did agree to opening up the salt dome at Gorleben and to having it tested to see if it would be suitable for final storage. In the meantime, shallow drilling in order to investigate groundwater movements in the surrounding rock and deep drilling for the investigation of the salt dome itself have been started. Despite this decision of the Government of Lower Saxony, the Federal Government is sticking to the integral waste management concept and is continuing to work towards its successful completion.

Reprocessing plant in Hesse However, in addition to the efforts towards setting up the National Fuel Cycle Centre, alternative solutions are also being sought. For example, the State Hesse has said that it will provide a site for a demonstration reprocessing plant with a capacity of 350 tonnes/year. An application for the construction of such a plant, which could be put into operation in 1990, was submitted to the appropriate State Ministry by the DWK in February 1980. Its throughput would be equivalent to one line of the large reprocessing plant; however, a series of steps has been taken to keep the radioactivity level as low as possible. Mainly, this is achieved through longer interim storage of the fuel elements, which is increased to 7 years instead of one year or less, as has been envisaged in the concept for the integrated fuel cycle centre [4]. Hence, reducing the storage capacity for spent fuel elements from 3000 t to 200 t of uranium the amount of deday heat in such a plant is reduced by a factor of about 100. Extending the cooling time to 7 years decreases the quantities of Ru-106 to about 1 % and of Zr-95 even to 0.1 ppb (10^{-10}), both nuclides which disturb the Purex-process. This obviates the need for interim storage of the highly active fluid waste, since vitrification can immediately follow the reprocessing stage. I.e. the removal of decay heat from the liquid waste tanks is reduced from 40 MW (fuel cycle centre) to less than 0.4 MW in the 350 t/y-plant.

The reduction in deday heat, both in the spent fuel element storage and in the highly active fluid waste storage by a factor of about 100 means an advantage from the safety engineering viewpoint and makes it possible to use passive cooling systems. Consequently, the term "inherent safety" is often heard in connection with the smaller scale reprocessing plant:- this requirement first emerged during the Gorleben symposium in respect of the stages "interim storage of fuel elements" and "HAW interim storage". It was then taken up in the findings of the State Government of Lower Saxony.

Interim storage of fuel elements in transport containers. The waste management strategy which resulted from the planning of the smaller reprocessing plant necessitates sufficient interim storage capacity in the form of so-called "central interim stores for fuel elements", to be built separate from the actual atomic power station sites. Initially, there are plans for 2 interim stores, in North Rhine-Westphalia and Lower Saxony, each with a capacity of 1500 tonnes of uranium. Both wet and dry stores are being considered, the dry stores being of particluar importance because of their "inherent safety" due to the use of purly passive components.

The dry storage concept [5]favoured by the FRG for fuel elements consists of the use of thick-walled cast steel transport containers with subsequent interim storage. Decay heat is dissipated via the surface of the containers by way of natural convection and thermal radiation. Depending on the type of fuel element, the capacity of the containers is between 2 and 3 tonnes of uranium, amounting to a total weight of 80 tonnes. Larger containers for holding up to 5 tonnes of heavy metal are under development (total weight 120 tonnes). The 40 cm thick container walls made of high ductility cast steel, and the mulitiple lid system, offer sufficient protection against any forseeable difficulties during transportation and storage. Thus, an interim fuel element store constructed around these transport containers would not require any additional safety precautions against external influences. The advantages of this type over conventional fuel element stores is the high degree of flexibility which it affords for extending the store as well as the facility for storing fuel elements actually on the sites of nuclear power stations, expecially where the construction of compact stores is technically not possible.

SAFETY PHILOSOPHY

Exposure to radiation in normal operation. Paragraph 45 of the Radiation Protection Order (Protection of the surrounding population) applies to plants in the fuel cycle just as it applies to nuclear power stations themselves. Within the framework of their consultations on the National Fuel Cycle Centre, the RSK and the SSK made the recommendations for the restriction of radioactive emissions in the exhaust air and waste water from the whole 1400 tonnes/year plant:

body area	exhaust air dose in mrem/y	waste water dose in mrem/y	dose limits mrem/y
whole body	4	2	30
thyroid	18	2	90
bones	12	5	90
skin	5	-	180
other organs	< 4	< 1	90

The radiation exposure of the personnel in fuel cycle facilities is reduced corresponding to the ALARA-principle (as low as reasonably achievable). Measured values for the demonstration reprocessing plant in Karlsruhe (WAK) show that the radiation exposure could be reduced considerably from the beginning of operation in 1971 until now. The mean values today are less than 10 % of the dose limits for radiation exposure (5 rem) which are laid down in paragraph 49 of the Radiation Protection Order. The collective dose during reprocessing of highly burned-up LWR-fuel is about 240 man-rem/GW_e.y [4].

Sources of interference The safety requirements placed on the individual stages of the waste management process are comparable with those on nuclear power stations themselves. There are, however, a few characteristic features which can be regarded favourably by comparison with nuclear power stations:

- the absence of systems to withstand pressure at high temperatures
- a considerably reduced coolant requirement

- no nuclear chain reactions
- the absence of short-life nuclides, such as inert gases and iodine-131
- longer reaction times in the event of disorders
- physical distribution of the activity level in disconnected process stages.

However, one disadvantage is the increased level of longlife fission products and atomic fuels.

The main sources of interference to be considered in the area of nuclear waste management can be listed briefly.

- The provision of a reliable dissipation system for decay heat - whether for fuel element storage or for the interim storage of highly active waste solutions - is one of the most important tasks. For although conventional processes (the wet storage of fuel elements and HAW storage in stainless steel tanks) do not pose any insoluble safety problems, particularly because in the event of a disorder, i.e. the failure of <u>all</u> the multiple cooling systems, days are available for suitable counter-measures to be taken. However, recent demands for "inherent safety" point to the abolition of longer term HAW intermediate storage and to the use of dry storage containers for fuel element storage.
- Fuel element stores should be designed to withstand external influences in such a way that they can survive interference and then be used safely again. This mean in particular that where the interference takes the form of an aircraft crash, the fuel element wet store must be housed in a bunker.
- In addition, all stages of the process where there are high levels of activity, such as the reprocessing plant, the HAW store, the vitrification plant, and parts of the Pu-processing system, require similar protection.
- Individual stages of the process must be of subcritical status so that two completely dis-connected events are always necessary before a critical situation is produced.
- The radiological affects of fires and explosions in the plant (red-oil explosions, oxy-hydrogen reactions, solvent or zircalloy fires, etc.) can be controlled by means of suitable filter systems (sand bed filters, deep bed fibre filters followed by Class S suspension filters).

Long-term safety The concept being considered in the FRG for the final storage of radioactive materials in a mine excavated specifically for this purpose in a salt dome is acknowledged internationally as the leading solution. The development of the North German salt domes is a well-known geological phenomenon which some 120 million years ago resulted in the formation of the Gorleben salt dome, at a time when Europe and North America still formed a single land mass. Since then, the form of the salt dome is practically unchanged.

Rock salt is particularly suitable for hermetically sealing from the environment any materials which are stored in it, especially if these materials give off heat, since

- it is free from open fissures because of its plasticity
- its high stability permits the excavation of relatively large cavities (without supporting structures)
- at temperatures up to 80°C its thermal conductivity is 2 to 3 times that of other rock types.

Permanent exclusion from the biosphere can, in the view of the RSK [1], be guaranteed by the use of the following barriers:

1) The geological salt formations selected for storage at a depth of several hundred meters will guarantee reliable exclusion from the biosphere, despite the thermal loads which occur.
2) Tight sealing of the storage chambers and bore holes will either prevent or limit any contact between lye and the radioactive materials in final storage in the event of the ingress of water or lye into the underground workings (this is conceivable only whilst the materials are being introduced into the store).
3) Moreover, minimising the leaching rates of the solidification products used and corrosion rates of the shell materials will ensure that only slight amounts of activity will enter the lye in the event of contact.
4) Finally, suitable measures (for example the design of the mine buildings, filling the shafts) will prevent any leaching radioactive materials from entering the biosphere by diffusion or convection.

The recently put forward concept of the direct final storage of fuel elements would, in the longer term, result in a clear increase in the activity introduced into the final store, since in this case, the quantities of uranium and plutonium entering the final store workings are about 100 times greater. Moreover, in the event of recoverable final storage of irradiated fuel elements, which is also under discussion, personnel will probably be exposed to increased doses of radiation because of the size of the units and the additional transportation which would be necessary.

R+D-programmes. RSK and SSK have produced a set of questions which they recommend to be submitted to the research and development [6] since the results will be necessary in the actual realisation of the individual parts of the centre. Furthermore the Hahn-Meitner-institute for nuclear research, Berlin (HMI) commenced the so-called "Projekt Sicherheitsstudien Entsorgung" (PSE), in orded to investigate faults and their consequences for individual steps in the back-end of the fuel cycle, i.e. transportation of spent fuel element, reprocessing, interim storage of fuel elements or waste-products, waste-conditioning and final storage.

In the past two decades (1960-1979), the Federal Minister for Research and Technology spent nearly one billion DM for investigations in the field of nuclear waste management. The present annual expense is 150-200 million DM. At present the industry spends about 1/3 of this amount, but in 2 or 3 years it will attain the governmental share.

CONCLUSIONS

The current waste management strategy, i.e. the interim storage of irradiated fuel elements in compact stores for subsequent reprocessing in foreign plants, can be viewed as adequate, at least in the short term. However, in the longer term, a strategy at national level is essential in order to reduce our dependence on other countries in the energy sector. There can be no doubt that the concept of the integrated Nuclear Fuel Cycle Centre represents the best means of realising this aim from a safety point of view. The method which, for political reasons, has emerged to date, must be regarded less favourably, because of the additional transportation which it involves. In my view, the direct final storage of irradiated fuel elements, especially in recoverable form, represents from the safety point of view, the least attractive of the waste management systems which have been discussed.

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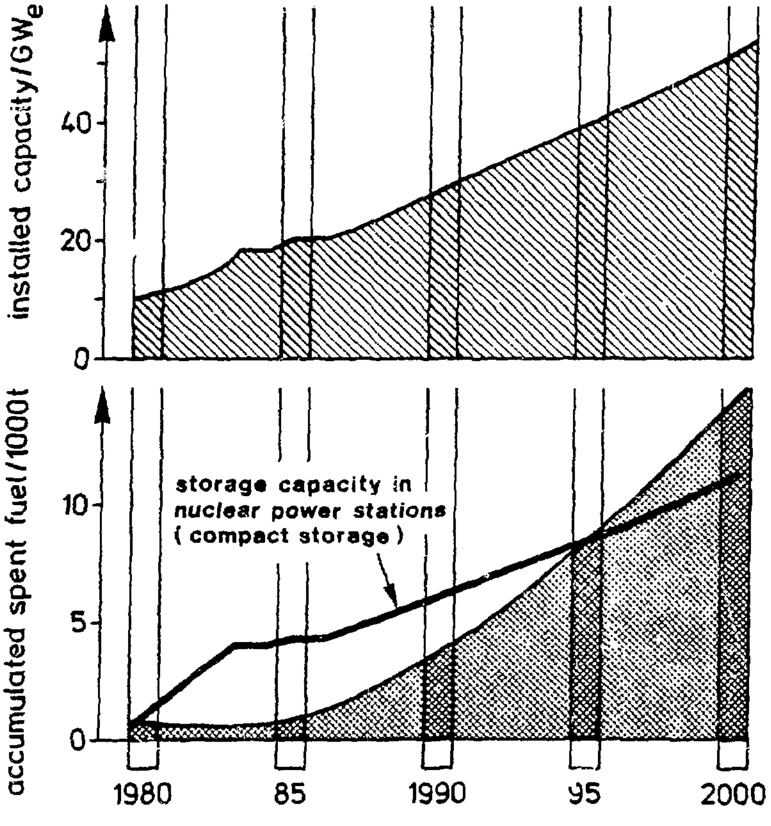
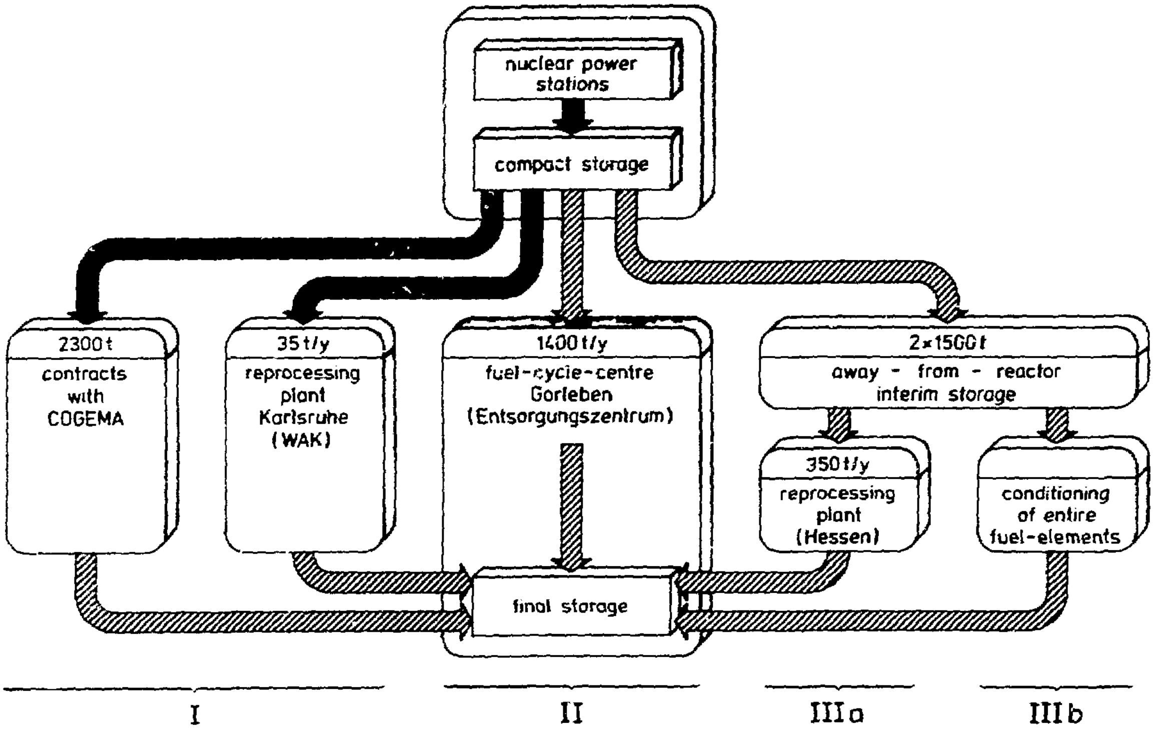
6. "Sicherheitstechnische Fragestellungen zum Entsorgungszentrum" GRS, Köln, Feb. 15, 1978

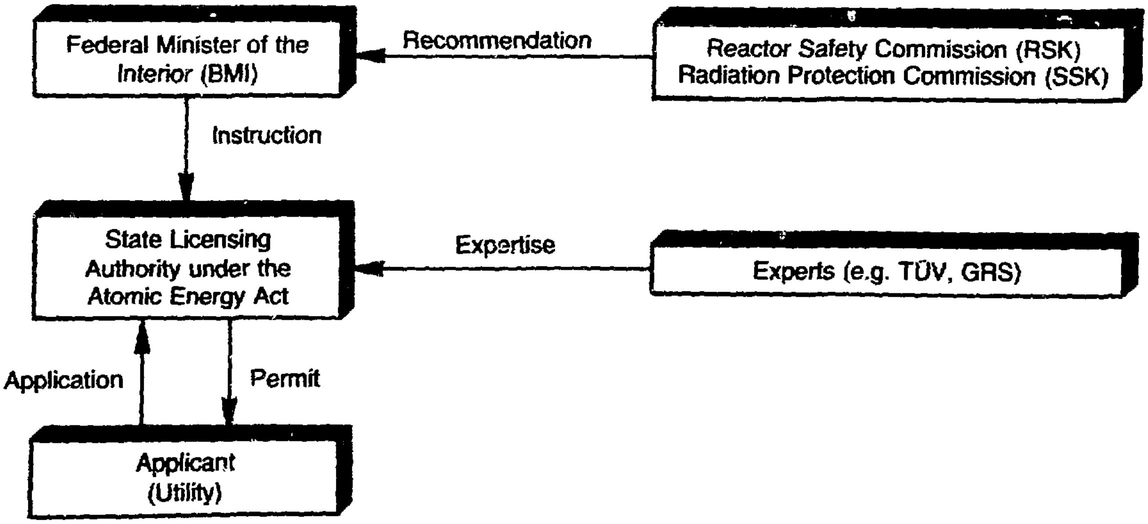
Accumulation of spent fuel from light water reactors in the FRG 1980-2000



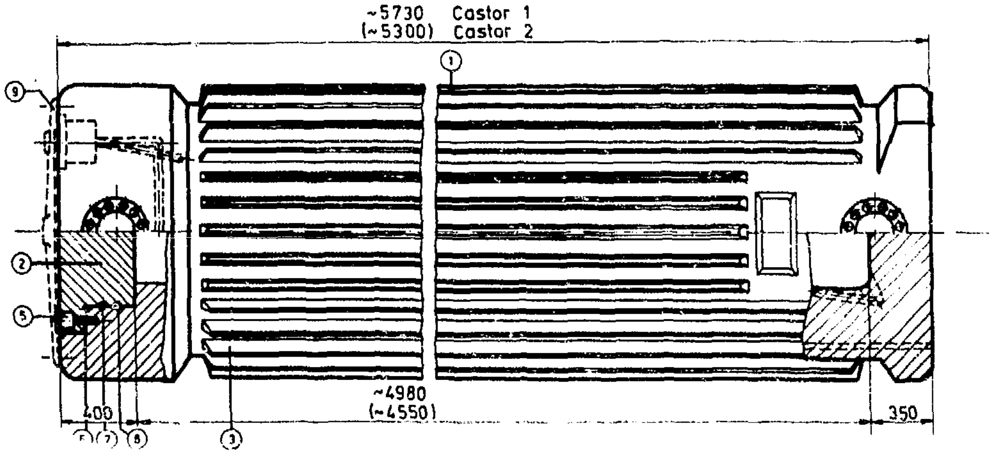
Concepts for nuclear waste management in the FRG

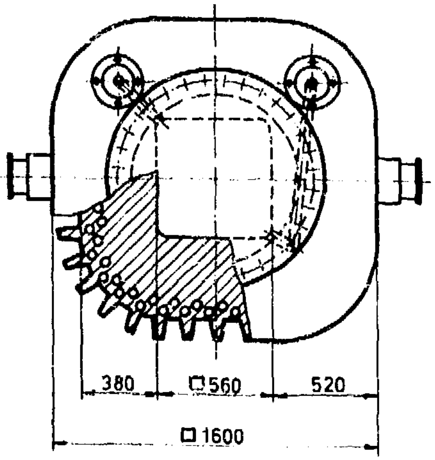


Licensing and supervision procedures



Transportation and storage of irradiated fuel-elements in cast-iron vessels



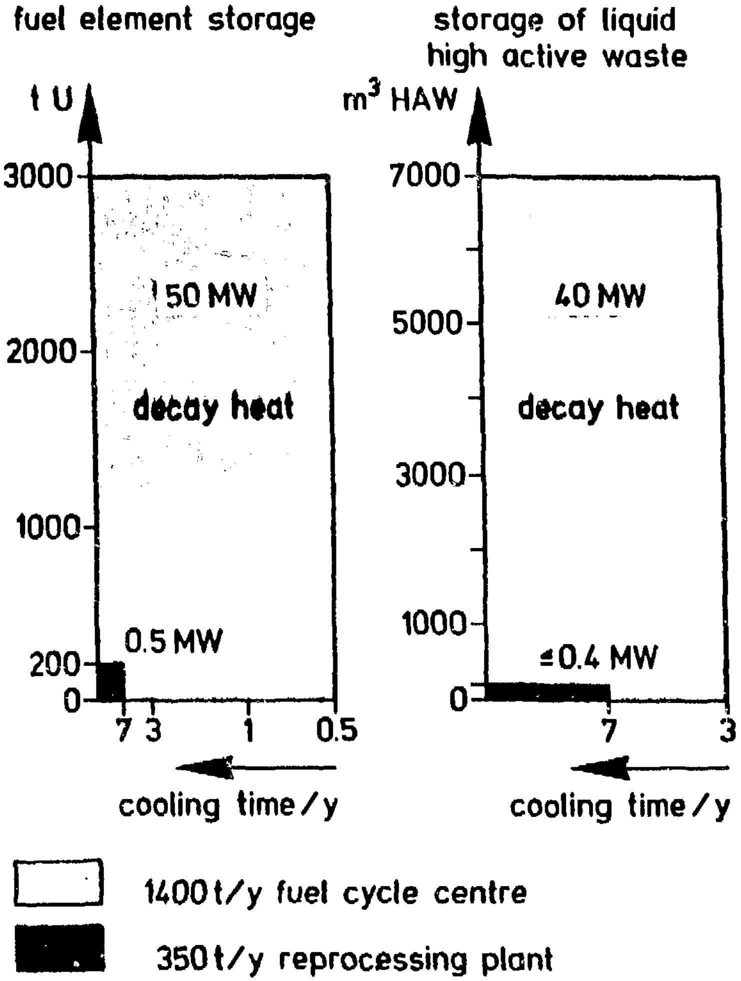


Castor 1:
4 PWR-fuel-elements
(2,1 t U)

Castor 2:
16 BWR-fuel-elements
(3,1 t U)

total weight: ~ 80 t

Optimization from safety engineering viewpoint



UNRESOLVED SAFETY ISSUES
WHERE DO WE GO FROM HERE?

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ABSTRACT

Section 210 of the Energy Reorganization Act of 1974, as amended requires the NRC to develop a program for resolving Unresolved Safety Issues related to nuclear power plants. Seventeen Unresolved Safety Issues were identified by the NRC in 1978 and by early 1979 the NRC Unresolved Safety Issues Program was quickly becoming a well defined and manageable effort. Although, the Three Mile Island accident caused the momentum developed in early 1979 to be lost, efforts on ongoing generic tasks were continued by a special NRC Task Force established in June 1979. The momentum that was lost must be regained, however, if the Congressional mandate in Section 210 is to be met. With increased industry involvement and the marriage of the Unresolved Safety Issues Program with the improved and broader safety program development, audit and evaluation activities of the new NRR Division of Safety Technology, this should be possible.

BACKGROUND

As a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended in December 1977 to include, among other things, a new Section 210 as follows:

UNRESOLVED SAFETY ISSUES PLAN

> SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter.

In response to the reporting requirements of the new Section 210, the NRC submitted to Congress on January 1, 1978, a staff report (NUREG-0410)[1] describing its program for resolution of generic issues. The generic issues program was already in place when the act was amended, but it was of considerably broader scope than the Unresolved Safety Issues Plan required by Section 210. Because of this, the letter transmitting NUREG-0410 to the Congress, indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It was the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on the most important issues, that is, those with potentially significant public safety implications. Thus in 1978, the NRC undertook a review of over 130 generic technical activities addressed in the NRC program to determine which issues fit this description and qualified as Unresolved Safety Issues for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

This selection process is described in a report (NUREG-0510)[2] transmitted to Congress in January 1979. The report provides the following definition of an Unresolved Safety Issue:

> An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants it affects.

Further, the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an Unresolved Safety Issue is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on those operating plants and plants under construction that are affected by that particular issue.

All of the issues addressed in the NRC program were systematically evaluated against this definition in 1978 as described in NUREG-0510. As a result, 17 Unresolved Safety Issues addressed by 22 tasks in the NRC generic issues program were identified. These issues and their associated generic task numbers are listed in Table I. Progress on these issues is discussed each year in the NRC Annual Report.

The NRC plans for resolving Unresolved Safety Issues are embodied in Task Action Plans that have been published and widely distributed in the past.[1,3] In February 1980, the most recent revisions of Task Action Plans addressing Unresolved Safety Issues were publised in NUREG-0469.[4] This document will be offered for sale by the NRC on a subscription basis in the near future.

Schedules for the tasks addressing Unresolved Safety Issues are displayed in one of the NRC's management and schedule tracking tools, the "rainbow books", in this case the Aqua Book[5]. The Aqua Book is updated quarterly and can also be purchased from the NRC.

TABLE I. "UNRESOLVED SAFETY ISSUES" AND APPLICABLE GENERIC TASK NUMBERS

1. Water Hammer (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity (A-3,4,5)
4. BWR Mark I and Mark II Pressure Suppression Containments (A-6,7,8,39)
5. Anticipated Transients Without Scram (A-9)
6. BWR Nozzle Cracking (A-10)
7. Reactor Vessel Materials Toughness (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolants Pump Supports (A-12)
9. System Interactions in Nuclear Power Plants (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment (A-24)
11. Reactor Vessel Pressure Transient Protection (A-26)
12. Residual Heat Removal Requirements (A-31)
13. Control of Heavy Loads Near Spent Fuel (A-36)
14. Seismic Design Criteria (A-40)
15. Pipe Cracks in Boiling Water Reactors (A-42)
16. Containment Emergency Sump Reliability (A-43)
17. Station Blackout (A-44)

TABLE II. STEPS IN RESOLVING AN ISSUE

STEP	PRODUCTS
1. Identify, Investigate & Evaluate Significance of Potential Issue	Decisions regarding: • Is/Is Not a USI • Priority • Need for Interim Measures
2. Plan Technical Approach, Resources, Schedule	Task Action Plan Aqua Book
3. Generate and Assemble Necessary Technical Information	Technical Reports
4. Evaluate and Decide What Licensing Requirements are Needed for Public Safety	NUREG Report Containing Proposed Requirements and Safety Evaluation
5. Peer, Public, ACRS and Industry Review	Comments
6. Promulgate Requirements	Orders, Letters, Rules, Guides, Standard Review Plans
7. Implementation	Changes in Design, Testing, Operation, Maintenance, Training, etc.

TREATMENT IN THE LICENSING PROCESS

In 1977, the Atomic Safety and Licensing Appeal Board in its River Bend decision (ALAB-444)[6] required the NRC staff to address in its Safety Evaluation Report for each application, "the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny." ALAB-444 clearly established that there must be a documented basis for concluding that there is reasonable assurance that plants can be safely constructed and operated with the answers to significant generic safety questions still pending.

The Appeal Board, however, left to the NRC staff the decision as to which generic safety issues were "significant" and therefore, had to be addressed in Safety Evaluation Reports. In answering this question the staff has equated the Unresolved Safety Issues reported to Congress with those that have "significant public safety implications" in the words of the Appeal Board. This was not done from the outset, but rather evolved over the course of several licensing proceedings.[7,8,9,10,11] Nonetheless, each of the "Unresolved Safety Issues" listed in Table I that is applicable to a particular plant being considered for a license is now specifically addressed in the staff's Safety Evaluation Report. A recent example of this is provided in Supplement 1 to the staff's Safety Evaluation Report for Sequoyah Units 1 and 2.[12]

RESOLVING AN UNRESOLVED SAFETY ISSUE

In the past, the resolution of an Unresolved Safety Issue has meant different things to different people depending on their perspective as a citizen, a regulator, a utility executive, or a Congressman. The basic steps in the resolution of an Unresolved Safety Issue which accommodate all these perspectives are given in Table II.

Steps 1, 2, 4, and 6 are performed by the NRC staff and its management. Step 3 can involve any combination of academia, industry organizations, and the NRC offices of Nuclear Reactor Regulation (NRR), Research and Standards and their consultants and contractors. Step 5 involves anyone who has a comment. With a few exceptions, comments are solicited on each of the NUREG reports providing the NRC staff's safety evaluation and proposed requirements published at the conclusion of Step 4. The advice of the ACRS is sought through interactions with the appropriate ACRS subcommittee at this point, as well as earlier in the process.

Step 7 must be accomplished by the owners of plants, together with their designers and manufacturers. Only after the hardware, procedures or training are changed to comply with the new requirements is an issue truly "resolved". However, it is convenient to track the decision point of Step 4 and the promulgation point of Step 6 as the times when the issue is "resolved generically", that is, when the staff (Step 4) and the NRC as an agency(Step 6) decide whether changes in current requirements are needed, and if so, what the changes should be. The NRC staff documents that have been published to date that constitute the product of the Step 4 or Step 6 decision for particular tasks are given in References 14-20.

THREE MILE ISLAND

In early 1979, the NRC staff's generic issues program was quickly becoming a well defined and manageable program. With the Section 210 mandate from the Congress, the NRC had come to grips with the question of which generic issues were the "significant" safety issues. This having been done, NRC management narrowed the focus of its manpower and contractual resources to concentrate on the list of Unresolved Safety Issues. Priorities were reordered and resources were being reprogrammed accordingly. For the first time, the Congress, the Commission, the Licensing Boards and the NRC staff were in general agreement on which generic safety issues were the most significant. This milestone was very important in defining and stabilizing the NRC's approach to considering generic safety questions in its licensing activities.

Then there was Three Mile Island.

The accident at Three Mile Island had a number of impacts on the NRC staff's Unresolved Safety Issues Program. The principal ones were that (1) it established quite clearly that the NRC's list of Unresolved Safety Issues was not complete; (2) it threatened to divert most NRC staff resources from the ongoing or planned tasks addressing Unresolved Safety Issues to Three Mile Island related activities for an indefinite period; and (3) it indicated that the approach being taken by the staff on several of the issues required some reconsideration.

To combat the problem of staff resources being diverted, a Task Force of about 30 professional NRC staff members was established in June 1979. The mandate of this Task Force was to continue work on those generic tasks addressing Unresolved Safety Issues that had previously been identified. The Task Force has been successful in keeping these generic tasks moving. Four NRC staff reports (Tasks A-9, A-12, A-24, A-42) have been issued for comment during this period and, although delayed, 4 more (Tasks A-2, A-7, A-10, A-36) are scheduled to be issued by the end of April 1980. Further, TMI impacts on the technical approach to solving the previously defined tasks were found to be minor and were factored in where appropriate. Neither the Task Force or any other NRC entity has yet, however, identified which of the Three Mile Island related generic safety issues should be added to the list of Unresolved Safety Issues for reporting to Congress and for addressing in Safety Evaluation Reports. It is planned to accomplish this task after the completion and approval of the TMI Action Plan.[13]

The Task Force approach has served its purpose, but its focus was narrow and its activities were peripheral to the TMI related activities that have dominated the agency's workers and managers for the past year. The momentum of the program for identifying and analyzing Unresolved Safety Issues and establishing and implementing new requirements was, in effect, lost. Substantial and rapid progress on resolving the backlog of significant safety issues was clearly the intent of the Congress in passing the legislation that amended the Energy Reorganization Act to include Section 210. For this reason the momentum of the Unresolved Safety Issues Program must be restored, and quickly.

WHERE DO WE GO FROM HERE?

Some very important steps toward regulatory recovery from the Three Mile Island accident have been accomplished. First, and most important, the safety of operating plants has been significantly improved. Second, the principal investigations of the accident are finished. Third, NRC has been through several iterations and refinements of a comprehensive plan for responding to the recommendations of the investigations.[13] This TMI Action Plan is nearing final approval. Fourth, a set of requirements within the TMI Action Plan has been developed and preliminarily approved for those operating license applicants whose construction is complete and who are otherwise ready to be licensed except for the TMI-related concerns. Having accomplished these steps, the NRC has made real progress towards getting its house in order, and the stage is now set for ending the licensing pause.

NRR resources will be concentrated for some time on those activities necessary to resume licensing and to assure the continued safe operation of plants. Nonetheless, there is more to be done at both the technical level and at the policy level to establish the course of regulatory safety policy and practices for the future. The technical projects that follow from TMI are described and arranged in the TMI Action Plan. In addition, the NRC has recently taken a major step in consolidating its functions of safety program development, audit and evaluation.

These functions have been recently established in a new division level organizational unit in the Office of Nuclear Reactor Regulation, as part of a broader reorganization soon to be effective, that will playa key role in implementing and continually updating the safety program for nuclear power plants. This division, the Division of Safety Technology, will be tasked with developing a more ordered, deliberate and reasoned process through which new requirements and backfitting decisions can be considered in the broader context of overall plant safety. In view of the fact that there is no generally understood or widely accepted national reactor safety objective, this will, as it has been in the past, be a difficult undertaking. Nonetheless, if regulatory uncertainty is to be reduced and regulatory actions understood by the public and the industry, significant strides in this direction must be made. The piecemeal approach to licensing requirements that has dominated reactor regulation for 20 years and has operated at an accelerated pace since the TMI accident, must be replaced by a more controlled and understandable one. The new Division of Safety Technology will have responsibility for doing this, while at the same time acting vigorously, with critical self examination of past practices, to overcome the complacency in reactor regulation so soundly denounced by all of the major investigations of TMI.

How do these changes in outlook and attitude affect the NRC's Unresolved Safety Issues Program? There are several ways. First, the Task Management function for Unresolved Safety Issues will be housed in the Generic Issues Branch in the new Division of Safety Technology. Second, the division will play the principal role in deciding which new issues or TMI follow on issues are Unresolved Safety Issues for reporting to Congress and addressing in Safety Evaluation Reports. A report on this subject is due to the Congress by July 1, 1980. Third, the division will establish the priority for resource allocation for these new issues. Finally, proposed new requirements that result from Unresolved Safety Issue tasks will be subjected to the improved review and approval processes to be developed by the new division. The steps in resolving an issue have not changed. However, the structure, the allocation of responsibilities and the individuals involved will change and the program's momentum will be restored.

Industry can and should play a key role in this process in the future. In the past, the identification and resolution of Unresolved Safety Issues has to a great degree been the business of the NRC staff. After all it was the NRC staff that was required by the Congress and the Appeal Board to improve its past performance in resolving generic safety issues. This had the effect of absolving the industry from assuming its share of this responsibility. The new Division of Safety Technology will seek new ways to substantially increase industry involvement in this process.

Notwithstanding the enormous demand for NRC staff resources for operating plants and near term operating license applications, the Unresolved Safety Issues Program still enjoys high priority attention among the large number of NRR activities. With some needed improvements in the administration of the program and its marriage with the improved and broader safety program development, audit and evaluation activities of the Division of Safety Technology, there is reason for optimism that the program can provide the necessary focus and mechanism for analyzing, evaluating and implementing the solutions of generic Unresolved Safety Issues in a timely and effective manner. This was the intent of the Congress and will be the goal towards which NRC will work as the TMI regulatory pressures subside and the new NRR organization develops and implements its new policies and practices.

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COST COMPARISON OF CENTRAL ELECTRIC POWER GENERATION USING COAL AND NUCLEAR FUELS

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ABSTRACT

This paper addresses the current and expected future costs of generating electricity with the two available practical modes of power generation, coal and nuclear. It describes the procedures and inputs used to arrive at the conclusion that generation with nuclear fuels will be about 16% more economical than generation with the best coal alternative.

Recognizing the uncertainty in long range estimates of this type, various sensitivity checks are developed to determine how much the capital, fuel, and operating costs would have to change to force a change in the ranking of the alternatives. The results are current estimates of the costs of generating electricity in the future in the middle western area of the United States with large nuclear units, and with comparably sized and comparably loaded coal units firing high and low sulfur coals.

INTRODUCTION

Two fuels will almost certainly be the sources of electricity for the next several decades. This was the recently reported conclusion of a four-year Department of Energy (DOE) sponsored study made by the National Academy of Sciences (NAS). The NAS Committee on Nuclear and Alternative Energy Systems (CONAES) concluded that "as fluid fuels are phased out of use for energy generation, coal and nuclear power are the only economic alternatives for large scale application in the remainder of this century. A balanced mix of coal and nuclear generated electricity is preferable to the predominance of either."

Before describing how we derived the figures in this paper and reached our conclusions as to the relative future costs of these two alternatives, several caveats and a discussion of the uncertainties involved are in order. First, the method used is that of obtaining the lifetime levelized future generating cost of each alternative. A levelized cost is a constant annual cost that is equivalent to the actual time varying annual cost when consideration of the time-value of money is included. This procedure is selected as a measure of merit because it is relatively easy to understand and is commonly used by the electric utility industry. In a site-specific comparison of the economics of coal versus nuclear generation, recognition would have to be made of the particular utility's requirements, capabilities, and external conditions in existence at the time. Generation expansion models would be used for planning simulations, and total system costs and reliability factors would be considered in detail.

UNCERTAINTIES

Much has been said about the uncertainties surrounding cost estimates of nuclear power generation. There is about as much uncertainty, however, in the economics of coal as there is with nuclear generation:

- Coal project durations have been extended to achieve compliance with air and water pollution regulations and the requirements for flue gas desulfurization equipment.

- A recent EPA report shows cost variation ranges of 2½ times in capital costs and five times in operating costs for flue gas cleanup.

- Standardization of coal units is hampered by the non-uniformity of coal burned in power boilers.

- Since 1969, costs of coal at the mine have increased an average of 15%/Yr. and coal rail tariffs have increased an average of 10%/Yr. If railroads are deregulated as proposed, tariffs applied to unit train coal movements may increase at even greater rates.

It is also true, however, that nuclear units face many serious uncertainties. For example, safety features prompted by Three Mile Island will add to the investment cost of new units. For every $10 million increase (in 1980 dollars) that is added to an 1100 MW nuclear unit, the generating cost would increase about 0.6%. Should the price of yellowcake increase $10/lb from today's level of $40, the lifetime average nuclear generating cost would increase about 3%. The yellowcake price would have to nearly triple while coal costs stayed constant, however, before levelized coal generating costs would become the more favorable. The enrichment price could go from about $100 to $255 per separative work unit (in today's dollars) before nuclear generating costs become less favorable than coal. If nuclear spent fuel disposal costs, including storage in a federal repository, encapsulation, transportation, and security, were to triple from the value given in a 1978 DOE report, the increase in levelized generating costs would be less than 4 mills per kWh or about 2%.

Much has been said about the economic uncertainty of decommissioning nuclear units, but this too has a relatively minor impact on overall nuclear generating cost. In this analysis, we have added a 30% contingency to the present day estimated cost of decommissioning, arriving at about 51 million 1980 dollars. Escalating this cost and then spreading it over the nuclear unit lifetime yields an addition to the nuclear generating cost of about 1.2 mills per kWh and increases the levelized nuclear generating cost by less than 1%.

A major risk faced with both coal and nuclear units is that of delay in the date of commercial operation of the unit. This is particularly serious in the case of nuclear units where investment costs are higher. For example, if an unplanned delay of two years should occur immediately after design engineering commences, the cost of the nuclear unit might increase more than $200 million. This could vary with the length of the delay, when the delay occurs and the interest and inflation rates prevailing at the time. It does not include the additional costs of replacement power during the delay, penalty charges, and costs resulting from escalating regulatory requirements during the longer schedule. Large financial risks such as these influence utility system planners and could lead to the abandonment of the nuclear alternative. This possibility could have disastrous consequences for the country.

PROCEDURE

Total generating cost is the sum of fuel costs, operation and maintenance costs, and the cost of capital investment in new generating plant. To combine capital costs with annual production expenses, we apply a percentage, called a fixed charge rate, to the capital investment cost. This factor annualizes the investment and permits its addition to the annual fuel, operation and maintenance costs. Financial mathematics are used to adjust for differences in the timing of the cash flows among alternatives. This is necessary when comparisons are made of investment in plant today and operating costs or savings in future years.

CURRENT GENERATING COSTS

Each year the Atomic Industrial Forum surveys electric utilities having both fossil and nuclear capacity in their systems. Forty-three of 48 utilities responded to the most recent survey. Of those reporting total costs, the average nuclear production cost for a kilowatt hour of electricity was 1.54 cents in 1978, about the same as in 1977 and 1976. A base load coal-generated kilowatt hour in these utilities' systems cost 2.15 cents in 1978, up from 2.0 cents in 1977 and 1.6 cents in 1976. These costs include fixed charges on plant capital, fuel, and operating and maintenance costs and represent the total cost of producing power up to the point at which electricity leaves the generating station. Although the savings from nuclear generation are not now as great as was envisioned in the 1950's, the nuclear generating costs represent savings of millions of dollars each year for America's electric consumers, compared to higher-cost generation that would have occurred with fossil fuels.

FUTURE GENERATING COSTS

The National Energy Act of 1978 prohibits the use of oil or natural gas as a primary fuel in any new base load generating plant, with some exemptions. However, even before the 1978 Act, fuel economics had driven utilities away from the use of gas and oil to generate electricity. About 60% of the new boilers ordered in 1971 were designed to burn oil or gas. The last such unit was ordered in 1974.

We will deal here only with base load bulk power generation using coal and nuclear fuels. Despite certain disadvantages with each, we must realistically acknowledge that these are the fuels to be relied upon for almost all of our country's electric power at least through the end of this century.

More exotic sources of energy such as solar and windmills are too far over the horizon to allow realistic appraisals of their future costs, reliability, and availability. Other sources of large scale power supply, such as geothermal and hydroelectric, are severely limited by geography. Only coal and nuclear fuels are now sufficiently developed to allow realistic cost comparisons for large scale electric generation.

RESULTS

The average annual generating costs projected for midwestern coal-fired and nuclear power plants over 30 years of operation, starting operation about 1992, are shown in Figure 1. For the conditions assumed, the 1100 megawatt-electrical (MWe) nuclear unit is expected to generate at 161 mills per kilowatt-hour (kWh), an average of 16% lower than that of the next most economical alternative, a comparably sized unit fired with high-sulfur coal from central Illinois. During the 1992-2021 period, nuclear costs are thus forecasted to average about ten times higher than they are today, primarily due to inflation. The average cost for coal generation, however, will be even higher, primarily because coal generation is affected more severely by inflation.

Explained below is how the various components of generating costs shown in Figure 1 are derived.

CAPITAL INVESTMENT COSTS

The capital investment in a power plant is made up of four elements - direct costs, indirect costs, an Allowance for Funds Used During Construction (AFUDC), and escalation.

DIRECT COSTS are those costs which would be incurred by the utility if it could purchase all equipment and materials and construct the generating unit instantaneously at today's price levels.

INDIRECT COSTS include capital charges to the utility company that are beyond the direct costs of equipment, materials, and construction labor. They generally include charges for consulting engineering, construction management, quality assurance, permits, and other costs incurred during construction.

AFUDC is a cost added to the cost of the generating unit to compensate investors for the use of their money during the lengthy period between the time funds for building the unit are spent and the time the unit goes into operation. If construction work in progress is not included in the rate base, electricity rates are not increased sufficiently to recover these costs for money obtained from stock and bond investors during the construction period. Instead, over the life of the unit, the utility will recover from its customers through depreciation charges, compensation for financing the investment made prior to operation of the unit.

ESCALATION

Suppliers of power plant equipment, materials and services usually link their prices to various statistical indices issued by the government and others.

One such commonly used index is the Bureau of Labor Statistics Standard Industrial Code (SIC) 36. Since the 1973 oil embargo and the end of price controls, average hourly earnings in the electric industry increased between an average of 7% and 11% per year through 1979.

The materials indices to which many power plant equipment suppliers link their prices have increased at even greater rates. A commonly used Bureau of Labor Statistics material index for metals and metal products, Code 10, increased almost 28% during 1974, moderated somewhat shortly thereafter, and appears once more to be increasing at a high rate, averaging over 13% annually in 1978 and 1979.

Other power plant building, material, and equipment prices have increased markedly since mid-1973. Turbine prices, for instance, have increased at a rate of over 10% per year, exceeding the general inflation rate over the same period of time.

Figure 2 shows that total electric utility construction costs in the northern midwestern area of the United States have increased more rapidly than has the Consumer Price Index in the last ten years and since 1973.

Figure 3 shows Sargent & Lundy's most recent estimates for the investment costs of nuclear and high and low sulfur coal-fired units going into commercial operation in 1991-92. The operating dates shown were chosen because we believe it would take a minimum of eleven years to design, license, construct, and test a nuclear unit from the time engineering is authorized, assuming a site and unit size have been preselected. Coal units can possibly be put into commercial operation about six to seven years after the engineer is authorized to start design activities. All units are assumed to be located in the north central part of the country to normalize the costs of construction labor and coal transportation.

Investment cost escalation resulting from expected inflation in the economy has been determined using a rate of 9.5% per year through 1981 and 9% per year thereafter. Escalation in the cost estimates is assumed to take place from now to the payment dates for material, equipment, construction labor, and services.

The rate used to add AFUDC to the escalated cost of equipment, materials, construction labor, and services is 9.5% per year, compounded semi-annually. This is compatible with AFUDC rates allowed today by regulatory authorities in many jurisdictions. Sales and use taxes have been excluded from the capital cost estimates.

An important and relatively recent consideration in the estimates is the inclusion of flue gas desulfurization (FGD) equipment with both high and low sulfur coal units. The New Source Performance Standards recently issued by the Environmental Protection Agency require such equipment, even with the use of low sulfur coal that is abundant in the western United States. Such FGD equipment adds between 10% and 20% to the present-day direct investment cost of each coal unit, not including the cost of sludge storage ponds for FGD system waste disposal. Also included in plant investment costs are closed cycle turbine exhaust cooling systems using mechanical draft cooling towers. Very few, if any, new large power plants in the U.S.

will be allowed to use natural bodies of water for condenser cooling due to the increasingly stringent environmental regulations in effect now and expected to be in effect in the future.

The total estimated cost of the nuclear unit, including all directs and indirects, escalation, and AFUDC, slightly exceeds three billion dollars, or $2,765 per kilowatt in 1992. Of this, about $793 million is the present-day direct cost of equipment, materials, service, and construction labor. These costs do not include plant modifications which might result from the accident at Three Mile Island. The NRC has identified some of these as including technical and operational support centers, control room redesigns for better controls and instruments, emergency power supplies for certain valves and indicators, additional provisions for isolation of the containment, post-accident radiation controls and plant shielding, greater use of reactor simulators, and other capital equipment. A final determination of the extent of these additional facilities, or even whether they are needed, has not yet been made. What is known is that these items and their associated engineering and construction will not be inexpensive. A measure of the uncertainty in these costs is reflected in NUREG-0660, wherein the NRC estimates them to be about $25 million and involve about 73 man-years of effort per unit. The AIF believes the cost of implementation could range from $28 to $700 million and could involve 100 man-years of effort.

FIXED CHARGES ON INVESTMENT

Utility revenues are largely controlled by regulation; planning in regulated industries is done by minimizing revenue requirements. Certain revenue requirements can be expressed as percentages of the capital investment and must be paid each year, independent of production from the generating unit. These so-called fixed charges include interest on debt and return on equity, and represent compensation to bond and stockholders for use of their money. Fixed charges also include Federal income tax, insurance premiums to cover non-nuclear related losses, state and local taxes and depreciation expenses. Nuclear insurance expenses are included in nuclear operation and maintenance expenses, discussed below.

Levelized fixed charge rates of about 18% per year are applied against capital investment in this comparison of generating costs. Discounting is based on a 10% per year projected cost of debt and a 10.4% rate for preferred stock. The projected yield on common equity is assumed to be 15% per year. The capital structure is such as to result in a weighted average return and discount rate of 11.9%.

FUEL COSTS

Fuel costs have increased sharply in the past several years, as shown in Figure 4. Oil has increased in cost by nearly a factor of three since the oil embargo in late 1973. Gas has gone up by more than a factor of four since 1973, but remains less expensive than oil because of more rigid price controls. The cost of delivered coal has more than doubled since 1973, mainly due to inflation, declining labor productivity, and higher transportation costs. Nuclear fuel costs have remained relatively stable, but large recent increases in natural uranium and uranium enrichment costs are beginning to be felt.

Low sulfur coal is assumed to come from the Powder River Basin of Montana and northeastern Wyoming. An average sulfur content of 0.5% by weight and heating value of 8100 Btu/lb are assumed, typical of coal from this region. Discussions with coal suppliers and utilities and examination of prices reported in the trade press indicate a current price, FOB mine, of about $8 per ton for long term contracts, including local taxes.

Transportation of low sulfur coal is assumed to be in 100-car unit trains traveling on a single railroad's line to the Midwest. Examination of published tariffs and discussions with railroad executives and utility users suggest that about $16 per ton is a likely estimate of the tariff a railroad might require for this transportation. Total delivered coal cost thus is approximately $24 per ton, or about \$1.50 per million Btu (MBtu) as burned. Escalation of the delivered coal price is estimated at about 9.9% per year, based on examination of historical relationships between mining and transportation costs and general rates of price increases in the U.S. economy. This forecast conservatively assumes a 7% long range general inflation rate and therefore assumes the delivered price of western coal will rise in constant dollar (real) terms at a rate of almost 3% per year. Further evidence of the conservatism of the coal prices used in this study is the existence of coal bids for slightly better quality western coal with prices considerably in excess of \$2.00/MBtu, including transportation cost.

High sulfur coal is assumed to come from central Illinois, containing 3% sulfur by weight and 10500 Btu/lb. FOB mine cost for this coal is taken to be slightly under \$23 per ton, including local taxes, and unit train transportation is estimated at \$6.80 per ton, for a total delivered price of about \$29.80 per ton or \$1.40/MBtu. This delivered price is forecasted to escalate at approximately 9% per year, again assuming that general inflation will be 7% per year in the long range future. Midwestern coal is forecasted to escalate less rapidly than Western coal because the delivered price of Midwestern coal contains less transportation cost, which historic data suggests may rise more rapidly than mining costs.

In projecting future nuclear fuel costs we have assumed a current yellowcake price of \$40 per pound of U_3O_8, which appears to be slightly above the price at which uranium now is marketed. Cost of conversion to hexafluoride is \$2.50 per pound of uranium, approximately the current market price of this service.

Enrichment processing by the U.S. Department of Energy (DOE) is priced in terms of dollars per separative work unit (\$/SWU). DOE's price of \$98.95/SWU, effective at the beginning of 1980 for recent enrichment contracts, has been used in these cost comparisons.

Costs of fabricating nuclear fuel assemblies are based on recent proposals made by reactor suppliers.

Each of the foregoing components is presumed to escalate at rates derived by comparing historic cost behavior with general rates of price inflation. The composite rate of escalation for nuclear fuel is about 9% per year, equivalent to a rate of 2% per year in excess of the rate of increase in the general economy.

Perhaps the most controversial aspect of nuclear power today is treatment of spent fuel assemblies. Studies by many government and private organizations support the position that radioactive wastes from reprocessing can be disposed of safely. In addition reprocessing appears to be desirable because of the resource conservation afforded. Recycling the uranium and plutonium recovered from reprocessed fuel is equivalent in energy content to over 30% of the newly mined uranium and over 20% of the separative work present in fresh fuel. Permanent disposal of spent fuel instead of reprocessing would deprive the economy of these resources. Present government policy does not permit reprocessing, however, so we have based our nuclear fuel cost estimate on "throwaway" fuel management, in which spent fuel is encapsulated and disposed of at a Federal repository. The nuclear fuel cost estimates assume a total cost of \$165/ kilogram of uranium in 1980 dollars for spent fuel transportation and disposal, based on escalation of DOE estimates presented in a July 1978 report.

OPERATION AND MAINTENANCE COSTS AND DECOMMISSIONING

Annual operation and maintenance costs for the coal units include staff labor, operating and maintenance materials and supplies, and administrative costs but exclude fuel expenses.

These costs are based on units equipped with mechanical draft cooling towers and limestone throwaway-type flue gas desulfurization (FGD) systems. Inclusion of FGD systems can add about 70% to O&M expenses for coal units constructed without such systems, reflecting the cost of large quantities of limestone reactant and the expense of FGD system waste disposal.

O&M expenses for the nuclear unit include additional security personnel, shift technical advisors, nuclear insurance, decommissioning costs and NRC inspection fees. The inclusion of a shift technical advisor is the result of a recommendation made by the NRC's Three Mile Island "Lessons Learned" Task Force.

Annual nuclear insurance premiums provide property, liability, and outage coverage. Government indemnity is not included because government coverage is expected to expire before the 1992 commercial operation date assumed here. As a result of Three Mile Island, the electric utility industry is in the process of establishing an insurance company, Nuclear Electric Insurance Limited, whose purpose will be to insure utilities against costs of prolonged outages and expensive replacement fuel. The annual outage insurance premiums of $1.5 million included in this analysis are based on maximum coverage of $156 million.

Annual escalation of 8.9% per year is assumed for labor, limestone, and sludge disposal expenses; 8.5% per year for maintenance materials; and 7% per year for NRC inspection fees. These rates are based on historical relationships between O&M cost items and general rates of price inflation as measured by the GNPD.

Decommissioning costs for the 1100 MW nuclear unit are based on recent industry studies. The prompt removal/dismantling method of decommissioning has been selected for this cost estimate. A 30% contingency and a 9.0% annual escalation rate have been included in the estimate of decommissioning expenses, which amount to a levelized value of 1.2 mills per kWh, representing a cost in today's dollars of about $51 million.

CAPACITY FACTORS

Capacity factor, the ratio of actual to maximum possible annual energy generation, determines the total energy production over which the fixed charges on capital investments are spread. The capacity factor depends on system demands, including assignment by the utility of generating capacity to meet demand, and on forced and scheduled outages.

Capacity factors of presently operating coal and nuclear units are comparable. An Edison Electric Institute study using data for the ten year period 1968 to 1977 inclusive, shows a capacity factor for all nuclear units of 61.23% compared with 58.35% for fossil units in the 400-599 MW size range and 56.53% for fossil units in the 600-799 MW size range. In 1977, nuclear plants operated with a 66% capacity factor, compared with 57% for coal units and 50% for oil units. In 1978, nuclear looked even better, operating at a 68% capacity factor versus 55% for coal and about 51% for oil units. Figures for 1979 were not in hand as of the time this paper was prepared, but the nuclear results will probably be less favorable because of TMI, and problems with seismic calculations which forced several units off line.

The economic comparisons presented in this paper for future units assume an average 60% capacity factor for both coal and nuclear units over their assumed 30-year operating lives. Operation at lower capacity factors would hurt the nuclear units because of their relatively high fixed charges. Operation at higher capacity factors, on the other hand, would make the nuclear units look better economically, since the fixed charges would be spread over a larger number of energy units (kilowatt hours).

SENSITIVITIES

Sensitivity checks have been made to determine how much the capital investment and operating costs of the nuclear unit would have to be increased before the nuclear unit might be less economical than the best coal unit on a lifetime evaluated cost basis. Under the assumptions cited here, the direct investment cost of the nuclear unit would have to increase 31%, or $247 million in today's dollars to reverse our findings. Spent fuel disposal charges would have to increase by approximately a factor of sixteen in today's dollars before coal became more economic. Alternatively, nuclear operating and maintenance costs (excluding fuel) would have to increase more than 2.5 times or decommissioning costs would have to increase 26 times in today's dollars to reverse the ranking. The capacity factor at which the coal and nuclear units operate in the future would have to be less than 40% before the coal unit became the economic choice.

Fuel and operating costs--those which will be incurred over the assumed 30-year lifetime of the units--represent about 66% of the total cost of the coal unit and only 41% of the cost of the nuclear unit. Therefore, continuing cost inflation will impact the coal alternative more severely than it will impact the more capital intensive but less fuel intensive nuclear unit.

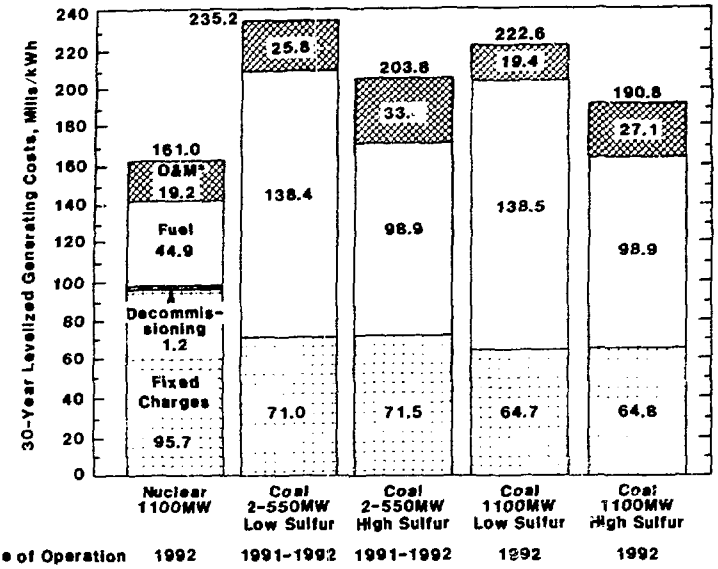
Actual data from operating plants demonstrates that nuclear generation is now more economic than coal generation. Our estimates lead us to believe that nuclear will continue to be more economic than coal in the future. Because of the distribution of investment and operating costs, nuclear power may also offer utilities greater long run protection against the effects of inflation.

CONCLUSIONS

Long development times are required for new forms of bulk power generation. It is neither prudent nor reasonable to believe that renewable energy sources such as solar power will be commerciallly available for bulk production of electric generation before the end of the century. That leaves only two practical and secure electricity generating alternatives for consideration for the near future -- coal and nuclear.

Today, nuclear power appears threatened. It has been beset by time-consuming and expensive regulation, uncertainties as to electricity demand, utility financial difficulties, and unusually rapid cost increases. Its public image has been damaged by a vocal and influential minority that has chosen nuclear power as a symbol of the establishment with which that minority is displeased. To this array of problems has been added perhaps the most serious of all: the accident at Three Mile Island. Should all of these concerns combine to make nuclear power infeasible, the American people will be the losers, for we will have reduced our practical and secure options for generating electric power from two - coal and nuclear, to one - coal alone. Coal may not be capable of bearing this load because of technical, environmental, legal and/or economic problems, thus forcing America to further increase its dangerous reliance on insecure foreign oil. If this happens, we will certainly see a serious decline in our standard of living as lights go out and factories shut down.

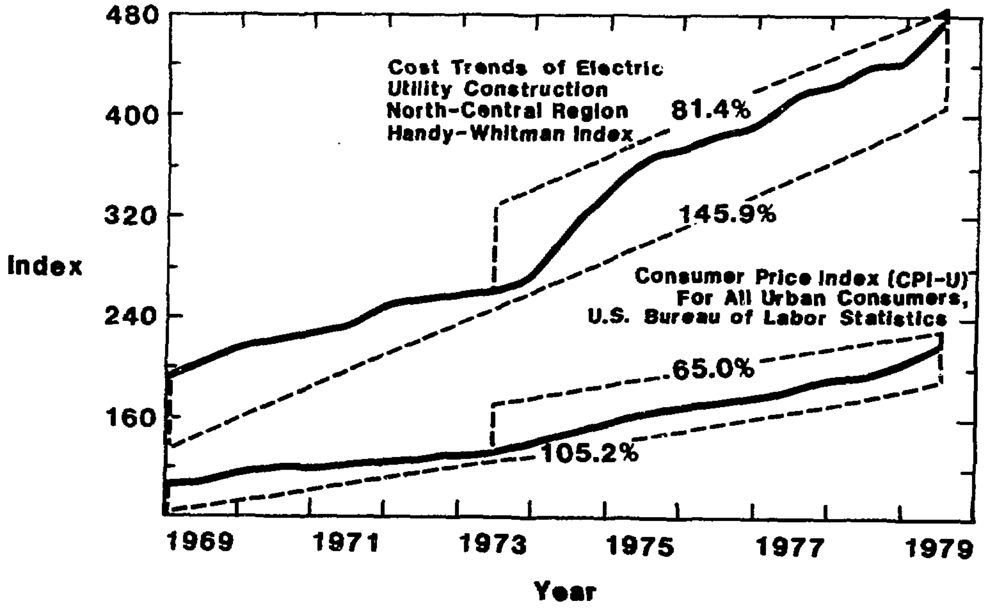
The United States is blessed with sufficient reserves and resources of both coal and nuclear fuels to carry us well through the period when the world's oil supplies are expected to run out. It defies all logic that we should be forced to depend upon unstable foreign governments, which can control our economic well being at their whim, for such a large share of our energy. We must ensure that circumstances will allow our nation to maintain a secure and adequate supply of electric energy. This can be achieved only if we base that supply on the two fuel resources, coal and nuclear, that are available <u>now</u>. We need <u>both</u> our coal and nuclear options to carry us through to the next century when yet-to-be-developed technologies might make meaningful contributions to our energy supply.



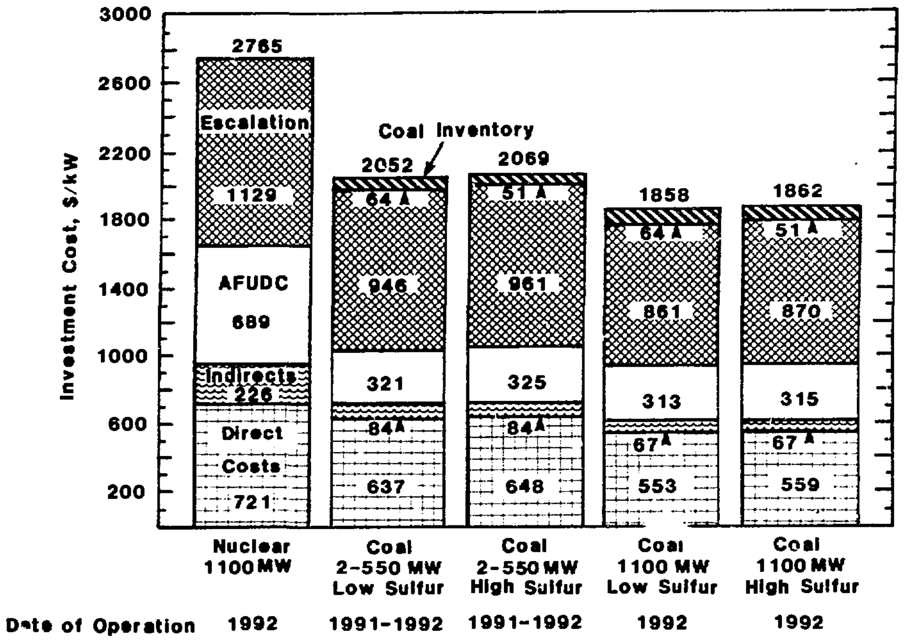
(60% Capacity Factor and Approximately 18% Fixed Charge Rate on Depreciating Capital and 21% on Fuel Inventory)

*Operation and Maintenance (includes nuclear insurance for the nuclear case)

30 - Year Levelized Generating Cost From 1992
Nuclear vs. Coal
FIGURE 1



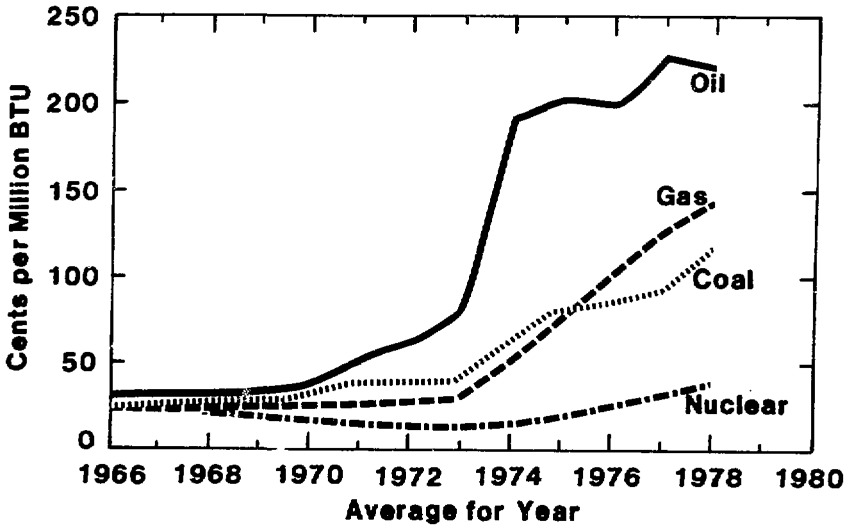
Electric Utility Construction Cost Trends
FIGURE 2



Source: Sargent & Lundy

Projected Capital Investment Cost Nuclear vs. Coal

FIGURE 3



Source: FPC (FERC), EEI, AIF, BLS

Fuel Costs of U.S. Electric Utilities (1966-1978)

FIGURE 4

THE FUTURE OF NUCLEAR ENERGY*

Alvin M. Weinberg
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In many ways nuclear energy is a fantastic success: a completely new source of energy now producing, or soon scheduled to produce, about 20 exajoules per year or almost 10 percent of all the energy man now produces. This energy will come from ∿ 525 large reactors in 36 countries. These reactors, if replaced by oil-fired power plants, would require about 10 x 10^6 barrels of oil per day — i.e., about one-seventh of all the oil produced in the world. Were the output of these plants used for electric resistive heating, in principle 2.5 x 10^6 barrels of oil per day could be displaced if in electric vehicles, perhaps 7 x 10^6 barrels.

Despite this extraordinary accomplishment, the first nuclear era seems to be coming to an end in many countries. Will there be a second nuclear era — that is, will nuclear energy occupy a secure niche as a large and permanent source of energy? Or will it simply be an ephemeral bridge to a fission-free future based on the sun, on geothermal energy, on fusion, and on fossil fuels — at least as long as the latter last, or until they are proscribed because of their effect on the climate?

It is impossible to generalize: in Austria, the first nuclear era has already ended or, more accurately, was not even allowed to start; in Sweden a majority voted to end it in 25 years; in the United States, some states have proscribed nuclear energy, and President Carter refers to it as an energy source of last resort. By contrast, in France, Japan, and the Soviet Union, nuclear energy continues to grow rapidly, and plans are going forward for the second nuclear era, based on breeders or other high-gain reactors.

The most plausible futures probably require nuclear energy. A world of 8 x 10^9 people is almost surely going to demand much more energy than we use today, assuming the energy can be found. R. Rotty of the Institute for Energy Analysis (IEA) and W. Haefele, et al. of the International Institute for Applied Systems Analysis (IIASA), visualize a world that uses 3-4 times as much energy in 2030 as we use now. Were most of this to come from coal, the world would have to mine 25 x 10^9 or more tons of coal each year. This I deem to be incredible. I would imagine the dangers of nuclear energy would pale by comparison.

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*Presented as the Closing Plenary Address, ANS-ENS Topical Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 11, 1980.

Yet a nuclear future of this magnitude is also formidable. Even if but one-half of this energy were produced by nuclear reactors, we would be speaking of a world of 7500 large reactors. Is this credible? In short, is a very large second nuclear era possible, even if the world allows the first era — based on reactors of current type and limited by the amount of relatively cheap uranium — to evolve into the second era, based on reactors that, in principle, can be supplied with fuel indefinitely.

That scenarios are uncertain goes without saying. At the recent Münster conference, Amory Lovins argued that improved efficiency in end use could assure the amenities we now enjoy in a world of 8×10^9 people using no more energy than is used now. This amounts to reducing the expenditure of energy per capita from 2 kW-years per year per person to 1 kW per person. I shall not be here in 2050, when this happy situation is expected to take place, so I shall never know whether Lovins will be proved right. Given the uncertainties, to proscribe the second nuclear era now on the grounds that the world can live in relative peace with an expenditure of 1 kW per person is mindlessly irresponsible. Nor can we count on the other options: each is beset with difficulties that all of us are familiar with. Nevertheless, no one can prove that nuclear fission is here to stay: our responsibility as nuclear technologists is to perfect the fission system so that it remains an available, politically acceptable option. Ultimately the future of nuclear energy is a political and economic question to whose resolution we nuclear technologists can only contribute, not decide.

II

I deal but briefly with the first nuclear era, during which nuclear energy is based on already developed reactors. Since a Pressurized Water Reactor, over its lifetime, requires about 6,000 tons of uranium, we had always understood that the first nuclear era was self-limiting. How self-limiting depends on how much uranium can be retrieved at an acceptable price — 10×10^6 tons would support 1,000 reactors for 50 years, for example. Thus we already may have in place 30-50 percent of all the reactors that will constitute the first era. What the ultimate usable price of uranium in current reactors might be is set by the price of energy from competitive sources. If the competition is, say, solar power towers, I suspect the upper limit for the price of uranium is far greater than we now imagine (though the world, paying so much for primary energy, would thereby be a far poorer place). If the competing source is the breeder, the upper limit might be, say, $180 per pound. (This is based on the breeder eventually costing $500 per kW more than the non-breeder.)

What can be done in the short run to ensure that the first nuclear era run its originally contemplated course is limited because the reactors and the institutions required to manage the nuclear enterprise are already in place. Two exceptions to this should be noted. First is waste disposal: a vigorous, clear demonstration of actual disposal of high level wastes would probably be as important as any single action to incline the public toward support of nuclear energy. Second, Three Mile Island may have proved, as Dr. Stratton explained at this meeting, that in accidents that develop slowly, the China Syndrome may be a myth: a melt-through with large release of radioactivity may be physically impossible. After all, it was the belief of the entire nuclear community, since 1960, that failure of ECCS would usually induce failure of the containment. If this is wrong, we must re-examine many basic assumptions. Moreover, I call your attention to calculations by S. Zivi that suggest the physical impossibility of the violent steam explosion blowing the top off a Pressurized Water Reactor vessel. These considerations, coupled with the observation that the ^{131}I source term may be grossly overestimated, represent the best of the good news from Three Mile Island. Nevertheless, even in the short run important fixes, though incremental, can and are being made.

Three Mile Island focused the public's attention on what many of us within the nuclear enterprise had realized was the real problem — the Class IX accident. That the enterprise has reacted vigorously — with the Institute of Nuclear Power Operations (INPO), the Nuclear Safety Analysis Center (NSAC), and insurance pools; and that a variety of technical improvements will be instituted can only be applauded. The aim must be to avoid another Class IX accident during the rest of the first nuclear era — not only because the public will hardly accept such an accident, but because, as the current moratorium on nuclear energy suggests, the financial strain on the affected institution is simply too great. No utility president is likely to order a nuclear plant if he believes he is betting his utility on an event (such as Three Mile Island) whose a priori probability might be as high as 10^{-2} per year.

Although reaction to Three Mile Island has not been uniform throughout the world, none can deny that its impact was profoundly felt everywhere. The U.S. utilities have recognized this in setting up INPO. Has the international nuclear enterprise reacted with equal vigor? Can we be assured that countries with little technological tradition can maintain and operate reactors safely? The industrialized countries have the strongest incentive to ensure that Class IX accidents are avoided anywhere in the world. The same considerations that led to establishment of INPO in the United States are relevant worldwide. Indeed, I would consider the extension of INPO, or something equivalent to INPO, worldwide as an extremely important step in ensuring that the first nuclear era is not aborted. I believe this matter is being taken seriously both by the International Atomic Energy Agency and by the Nuclear Regulatory Commission. *(I recall, during a visit to Pakistan in 1962, discussing with Francis Perrin the capacity of underdeveloped countries to manage nuclear systems. We were encouraged by the successful operation of national airlines in most of these countries: a handful of expert pilots and mechanics are sufficient to operate safely. However, as Three Mile Island has shown, once a Class IX accident occurs, the demands on the technological community become very great — much greater than can be met by the resources of all but the most sophisticated countries. Perhaps emergency response teams, combining international experts as well as experts from a nation's nuclear energy laboratory, ought to be established to prepare for such contingencies.)*

III

Are the measures now being taken to assure the continuation of the first nuclear era sufficient to ensure the second nuclear era: the era we visualized as involving perhaps 10 times as many reactors as we now have, many of these being breeders? Again, no one can tell; nevertheless, the argument used by the Swedish aeronautical engineer, Bo Lundberg, in 1963 with regard to the future of air transport, must be heeded. Lundberg pointed out that as the number of passenger-miles flown increased, the probability of accident per passenger-mile would have to decrease correspondingly. Otherwise the accident rate would increase — in his estimate, to several major crashes per day by 2010. Though the probability of a passenger successfully completing a flight remained as good in 2010 as in 1963, in Lundberg's view the public would lose confidence in air travel, and commercial air travel would collapse. He proposed that in the fifty-year period from 1960 to 2010, the fatality rate per passenger-mile would have to diminish from $11/10^9$ passenger-miles to about $.3/10^9$ passenger-miles — a factor of about 40.

Commercial air travel has actually become much safer per passenger-mile — so much so that although the passenger-miles have increased about as he predicted, the absolute accident rate has fallen. Over the last 20 years in the United States, though the number of active commercial transports has increased 19 percent, the total accidents have decreased 71 percent. By contrast, there has been much less improvement in general aviation: the number of fatal accidents in general aviation has doubled as have the number of airplanes. Yet the public tends to view accidents in small planes very differently than it does accidents in commercial transport. The 1270 people killed in 1979 in the United States in many small plane crashes would not be tolerated if they were killed in 10 or 15 large crashes each year.

The experience in air transport should teach us two things: first, that accidents do tend to diminish as experience is gained; but second, that as far as the public's perception is concerned, risk is not simply the product of probability x consequence — i.e., the first moment of the probability distribution of severity of accidents. Somehow, the public accepts many small airplane crashes, but reacts much more violently to a few very large crashes, although the total casualties are the same in the two instances. I would guess that this reaction is at least in part attributable to television: each of us can identify with, and be scared out of our wits by, a large accident that we see in detail on the TV screen. That the accident is a priori extremely improbable is less evident since what we see is an actual instance of the improbable occurring. In short, the public, I would suggest, understands consequences; it does not understand probabilities.

The a priori mean probability of Three Mile Island, according to Rasmussen, is about 4×10^{-4} per reactor year, with a tenfold spread on either side of this mean. If the first nuclear era amounted to 30,000 reactor years, and the mean a priori probability remains 4×10^{-4} per year, there would be, on average, 10 Three Mile Islands over the next 30-50 years, with the range lying between 100 and 1. This I would judge to be intolerable — not merely because the public would lose confidence in nuclear energy long before the tenth Three Mile Island, but because utility executives, whether private or public, would have lost confidence in nuclear energy. To survive the first era, I would suggest that we must reduce the a priori probability of Class IX accidents by a factor of the order of 10 to 100, so that at most there would be very few — say one or two Three Mile Islands, within this period.

IV

I shall not try to describe the many possible measures that can be undertaken to reduce the probability of Class IX accidents, or to mitigate their consequences. Many of these have been discussed at length at this meeting. They include a variety of technical and institutional fixes, mostly incremental. (For example, I have already implied that more careful analysis might rule out containment failures that are now conceded to be physically possible.) The possibility that has not been discussed is the development of reactor systems that are intrinsically less sensitive to meltdowns than are the present types. We are convening a small group of old-timers in the nuclear business (that is, the now rather elderly group of people who were responsible for setting the enterprise along its present course) to discuss whether the current moratorium in the U.S. might be used to advantage to establish criteria that reactors for the second nuclear era ought to meet.

One conjecture that I would put forward is that siting policy itself may have an influence on accident probability. If one concedes, as was assumed in the Rasmussen report when it admonished its readers not to multiply accident probability at time T by total reactor-years at time T + t, that the accident frequency per reactor per year diminishes as the total number of reactor years increases (according to the so-called cumulative learning curve discussed at length by P. C. Roberts of the United Kingdom), then it seems plausible to me that such learning occurs faster on a large site than it does on a small site, and that, therefore, the accident rate per reactor ought to be smaller on the larger site. To take an example, elements of the Three Mile Island sequence occurred at Davis-Besse, Oconee, and Rancho Seco before it occurred at Three Mile Island. Had all four reactors, Davis-Besse, Oconee, Rancho Seco, and Three Mile Island, been co-located, I cannot imagine Three Mile Island occurring. The word would have got around about the ambiguity in determining water level after a small LOCA. To be sure, INPO's and NSAC's main jobs are to ensure that the word gets around — i.e., that accident frequency diminishes fast enough as cumulative reactor years increase to more than balance the increase in number of reactors. I suggest that consolidation of siting would hasten the process, and thus ease INPO's and NSAC's task.

The trend toward consolidating siting is unmistakable: of the 525 reactors, 170, representing one-half the world's nuclear power outside the United States, are now on sites with 4 or more reactors. If this trend continues, then could we not contemplate a world of 5,000 reactors confined, say, to no more than 500 or 1,000 sites? Now if the cumulative learning curve for a multi-reactor diminishes so fast that the probability of accident per site is rather independent of the size of the site, we would be confronting a world in which the overall accident rate is not so different from what we now experience. I am able to contemplate such a second nuclear era with much more equanimity than I can one in which many thousands of reactors are scattered among very large numbers of organizations and sites, and in which the learning rate is correspondingly slower.

I realize that what I have said is conjecture. I put it forth for consideration; I should think that the influence of number of reactors per site on accident rate could be estimated from an analysis of LER's that are already available. This I should think would be useful datum to collect.

V

Much of Western society seems today to be afflicted by an environmental hypochondria that undermines and debilitates every massive technology. Is it possible that this hypochondria will pass, and that the public reaction to nuclear energy will eventually be commensurate with its true risks?

I see two possibilities. The first is that we will eventually be heeded in our insistence that nuclear risks must be judged in comparison to other risks. To take an example, Henry Hurwitz of General Electric has estimated that if the government's goal for conserving energy by better insulation of houses is achieved, then we can expect 20,000 additional lung cancers per year because of the increased exposure to radon in the tighter houses. This estimate is based on a strictly linear dose-response, with no threshold. The expected number of casualties during the next 20 years from insulating homes is therefore much larger than the casualties caused by the very worst Class IX accident that might occur in 20 years. Tightening houses to save energy is more dangerous than is a Class IX accident!

One cannot ignore Hurwitz's calculation: if the public reacted in a way that we here would deem rational, its fears about nuclear energy would surely be allayed by this argument. But the difficulty is the one I have already alluded to: a single incident that might harm many people is far more threatening than are many small incidents that in aggregate affect even more people. Nevertheless, I am optimistic enough to hope that people will eventually place risks of nuclear energy in perspective.

The other possibility is that the estimates of the amount of cancer caused by low levels of radiation could prove to be greatly exaggerated. The large number of cancers supposedly caused by the worst Class IX accident occur mostly among a very large number of people exposed to less than 1,000 mr per year of radiation — i.e., 3 mr per day. If low level radiation could be shown to be much less harmful than is suggested by the usual linear hypothesis (with a slope of 1 cancer per 5 x 10^3 rads), then the spectre of a reactor accident conceivably causing hundreds of thousands of casualties would be extirpated.

Three recent findings bear on this all-important issue. First, in the April 4, 1980 issue of *Science*, Raabe, Book, and Parks have shown that at least for bone tumors caused by radium, there is, in fact, a practical threshold — i.e., the latent period for appearance of the tumor exceeds the life span if the dose is lower than 39 millirem per day. This evidence is consistent with the findings at Nagasaki where low LET radiation below about 50 rads showed no increase in leukemia (even though the exposure was instantaneous); it is not consistent with Hiroshima data where there was a higher irradiation by high LET radiation and linearity perists below 50 rads. It is significant that the third

BEIR report of the National Academy of Sciences no longer accepts linearity below 10 rads — yet most of the 45,000 estimated number of cancers from the worst Class IX accident are attributable to lifetime doses less than 10 rads.

A second possible misconception is the alleged sensitivity of the fetus to prenatal radiation. One of the more dramatic events at Three Mile Island was Governor Thornburgh's order to evacuate pregnant women. The scientific basis for this action lies in the claim by Stewart and Kneale that the doubling dose for childhood cancer is less than 2 raas to the pregnant mother. This claim has been in the literature for about 20 years; it has been a source of dispute ever since it was made. During the past year Drs. J. Totter and H. G. MacPherson of the Institute for Energy Analysis have found a methodological flaw in the Stewart-Kneale analysis: namely, that the controls did not in fact match the cancer cases in many essential respects. In particular, the requirement that the probability that a control received X-rays equal the probability that a non-radiogenic cancer received X-rays was not fulfilled; as a consequence, the findings of Stewart-Kneale were rendered invalid. There is, according to MacPherson and Totter, no evidence that extremely low levels of prenatal radiation increases the probability of childhood cancer.

Finally, I call to your attention the recent article in the *Proceedings of the National Academy of Sciences* by John Totter on the origin of spontaneous cancer. Totter first shows that mortality from cancer, when corrected for competing risks, seems to be independent of a country's state of industrialization, and therefore of its level of man-made pollution. Thus, he argues, one must seek the primary carcinogens not among man-made agents, but rather among all-pervasive "normal" components of the environment. The culprit suggested by Totter is oxygen. His main argument rests on the known fact that one intermediate in the metabolism of oxygen is the superoxide radical, O_2^- ; and this radical is essentially the same as the radicals produced by radiation, which of course is known to be a carcinogen. Indeed, the radiomimetic dose continually imposed on each of us because of the flood of O_2^- radical might be between 500 and 2,000 rads per lifetime — i.e., between 7 and 30 rads per year; it is this flood of radiomimetic radicals that, in Totter's view, is an underlying, perhaps the most important, cause of cancer. If one accepts Totter's view, then the lifetime dose of 7 rads of background radiation, even on the linear hypothesis, would account for about one-third to 1 percent of cancer.

It is too early to say how Totter's revolutionary theory on the origin of cancer will be received by the scientific community. Thus far it has been promoted by the President of the National Academy of Sciences, Dr. Philip Handler who, along with Professor Fridovitch of Duke University, discovered the enzyme super-oxide dismutase that protects us from this enormous natural flood of radiomimetic radicals. Nevertheless, the evidence pointing to oxygen as the culprit is tantalizing: oxygen is known to be a mutagen; it has been shown to cause tumors in fruit flies; and it gives a positive Ames test, the assay that is often used to screen carcinogens.

I cannot say where these considerations will lead. I would suggest that they may very well result in our realizing that in fact low-level radiation is far less damaging than even the linear hypothesis suggests, and that therefore most of the fears concerning the lingering effects of Class IX accidents or, for that matter, of conceivable contamination from leaks from waste depositories, are unfounded. If these speculations prove correct, then I should think the Western world will come to its senses with respect to nuclear energy.

I close by drawing from William Clark's perceptive paper on "Witches, Floods, and Wonder Drugs." He likens the current environmental hysteria to the fear of witches that swept over much of Western Europe and America in the 16th and 17th centuries. The symptoms were much like those we now see every night on TV: vague discomforts, cattle dying, babies deformed because of industrial miasmas. Perhaps most striking was the hysterical fear exhibited by 400 Middletowners when the Nuclear Regulatory Commission proposed to vent 60,000 curies of ^{85}Kr from Three Mile Island: the maximum beta skin dose per person would have been 11 mr, the whole body gamma dose 0.2 mr (compared to Totter's estimate of radiomimetric O_2 dose of between 7,000 and 30,000 mr per year). Witch hunting flourished for two centuries, especially since it was in the interests of the witch hunting profession to find and burn more and more witches. It was not until 1610 that the chief inquisitor, Alonzo Salazar y Frias, became suspicious that the alleged connection between witchery and human ills may have been exaggerated. He ordered an investigation and discovered that although more than 500,000 bona fide witches had been burned at the stake in the past century, nothing else seemed to have changed: people got sick and died, wars and pestilence abounded, crops would sometimes fail. Though he did not proscribe witch hunting, he forbade the use of torture to extract confessions: the result was that witch burning, and then witch hunting, fell precipitously.

I do not wish to leave the impression that a Class IX accident is as innocuous as witches have turned out to be: we know that the LD_{50} is 400 rems of radiation and that in the worst conceivable accident some acute deaths would occur. But we also know that most of the presumptive casualties and the fear of Class IX accidents comes from low level exposure. I would therefore insist that the future of nuclear energy, whether there will be a second nuclear era, will depend upon the public's overcoming its unreasoning dread of our modern witch — exposure to low level radiation. It took the Inquisition more than a century to overcome its fear of witches. I would hope we will lay to rest this modern witch soon enough to ensure that the first nuclear era run its course, and the second nuclear era be allowed to co-exist with the solar era or fusion era.

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