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PRISM™

Preliminary Safety Information Document

Prepared for U.S. Department of Energy
Under Contract No. DE-AC03-85NE37937

Volume V

Appendix F

APPLIED TECHNOLOGY

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Letter dated 5/26/93

GENERAL ELECTRIC
ADVANCE NUCLEAR TECHNOLOGY
SAN JOSE, CALIFORNIA

AMENDMENT 9

87-568-05

DISCLAIMER

Department of Energy
Washington, DC 20585

MAY 26 1993

Mr. Dennis M. Crutchfield
Associate Director for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Crutchfield:

As you indicated in your letter, dated April 29, 1993, you are completing the final Preapplication Safety Evaluation Report (PSER) for the "Power Reactor Innovative Small Module" (PRISM) Advanced Liquid Metal Reactor design. You expressed concern about meeting one of the Commission's objectives of public disclosure since the PSER will be based on documents on which the Department of Energy (DOE), Office of Nuclear Energy, placed a restrictive distribution labeled "Applied Technology." We hereby approve your request for public disclosure and you are authorized to remove the "Applied Technology" (AT) distribution limitation from all of the DOE documents titled Preliminary Safety Information Document. The documents are:

"PRISM - Preliminary Safety Information Document" (PSID) - GEFR-00795

Volume I - December 1987, Chapters 1-4
Volume II - December 1987, Chapters 5-8
Volume III - December 1987, Chapters 9-14
Volume IV - December 1987, Chapters 15-17 and Appendices A-E
Volume V - February 1988, Amendment to PSID
Volume VI - March 1990, Appendix G

With regard to the Modular High Temperature Gas-Cooled Reactor (MHTGR), we would like to request that public disclosure of its AT information be delayed until publication of the MHTGR PSER becomes more imminent. We would appreciate your understanding of this situation and assure you that we will release MHTGR AT for public disclosure when needed to support the PSER issuance. We will be happy to meet with you and your staff to discuss this further at your convenience.

Sincerely,

Jerry D. Griffith

Jerry D. Griffith
Director
Office of Advanced Reactor Programs
Office of Nuclear Energy

cc:
Salma El-Safwany, DOE/SF
James Quinn, GE
Richard Hardy, GE
Robert Pierson, NRC
Ray Mills, PDCO

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ABSTRACT

This document is a Preliminary Safety Information Document (PSID) for a PRISM (Power Reactor Inherently Safe Module) electric power plant. The PSID is the document in the PRISM licensing plan that provides the description and evaluation of the conceptual design using nine reactor modules. Each module is a compact liquid metal reactor of the pool type design. The reactor module has unique passive safety characteristics that enhance the safety of the design. These include passive shutdown heat removal and passive reactivity shutdown. The document presents design criteria, design description and analyses that demonstrate these favorable safety characteristics. The format is similar to the standard format for safety analysis reports, however, the design description and evaluations are consistent with the conceptual design level. Design basis accidents are described in Chapter 15 and a preliminary PRISM probabilitic risk assessment is included in Appendix A.

RESPONSES TO NRC COMMENTS

APPENDIX F

RESPONSES TO NRC COMMENTS

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ABSTRACT

DOE requested a formal NRC review of the PRISM PSID on November 17, 1986 with a completion at the end of CY 1987. The review is followed by a final Safety Evaluation Report (SER) and licensability statement in April 1988. The purpose for this Volume V is to retain the GE responses to the series of NRC review comments on selected PSID chapters throughout 1987.

Volume V is titled Appendix F, Responses to NRC Comments. The Table of Contents for this appendix is structured by PSID Chapter 1 through 17 and Appendix A through E. Responses and pages are consecutively numbered for each chapter and appendix. Each response will follow a repeat of the NRC comment. Each page of Appendix F includes the respective amendment number for a change to the PSID. Page assembly instructions accompany each amendment.

This retention of the responses to NRC comments provides correlation between PSID section and NRC comment.

1.1 Comment

The PDCs described in Section 1.2 are quite general and broad. Our advanced reactor review policy requires that we build on existing criteria and methodologies. Based on the information you provided in Section 3.1 of the PSID it appears that many of the GDCs for LWRs from Appendix A to 10 CFR 50 have been applied to your design. We believe you should indicate this in Section 1.2. We also believe that you should use the criteria that have been developed by the ANS 54.1 Committee or those applied to the CRBR, where practical. These should be supplemented, where necessary, by criteria specific to PRISM to generate a set of PDCs for the PRISM plant being reviewed. (See 3.1 below).

Response

The PDC's GE provided were the top level ones used to develop the conceptual design of PRISM. It is true that as design details have developed more detailed criteria have been established and used in the design. These lower level criteria tend to be more design specific and prescriptive and may also change as the conceptual design evolves. It was for these reasons that we did not include lower level criteria such as 10CFR50, Appendix A, ANS 54.1 or CRBRP in Section 1.2.

In response to your comment we have expanded and included some of the more detailed criteria of 10CFR50, Appendix A and the CRBRP PDCs. Only those criteria that we currently satisfy and intend to continue to satisfy throughout the evolution of the PRISM design and that we judge provide useful guidance and are not redundant have been included in the expanded set of PDCs. Criterion 1 from 10CFR50, Appendix A has been omitted from this revised set of PDCs because it is largely a procedural requirement that we intend to meet as shown in Section 3.1 and through our more detailed discussion in Chapter 17.0. It does not appear to qualify as design criterion in the sense of specifying the design and therefore was not included in our expanded set of PDCs.

1.2 Comment

Since you intend to minimize the role of evacuation for this plant we believe you should include PAG dose guidelines in the consideration of containment performance requirements discussed on page 1.2-4.

Response

The emergency planning analysis of Chapter 13 which follows the guidance provided in 10CFR50, Appendix E, and the underlying technical basis for NUREG-0654-RI-1980 provided in NUREG-0396; shows from the PRA (Appendix A) sensitivity study using conservative accident source terms that, for PRISM, evacuation and sheltering are not significant to the control of public risk. As PRISM analyses advance we see more agreement with these results. As we understand the PAGs, they have been established for use in real time accident situations. Their role as a design criterion is not clearly defined. We believe the established practice of using accident assessment combined with sensitivity analysis of the risk reduction benefits of specific evacuation planning does not require the use of PAGs. Further, it would appear to add redundant requirements to the containment to follow the suggestion in your comment therefore no change has been made on page 1.2-4.

1.3 Comment

Please provide summary descriptions of the development tasks listed in Table 1.5-1. In addition, please consider including in this table some of the programs discussed in Chapter 14, particularly appendices 14A and 14B. Please address specifically the need for critical assembly tests.

Response

Summary descriptions of the development tasks have been included in a revised version of PSID Chapter I, Section 1.5. The development tasks have been identified for the complete certification process including the prototype reactor safety tests. It should be noted that some of the systems and components qualification tests will be performed in the prototype reactor prior to the safety tests. The key objective of the safety test is to validate the inherent safety characteristics of the reactor system including inherent reactor shutdown and shutdown heat removal. Specific instrumentation, as required, will be identified and qualified (see PSID, Chapter 14) for these tests. Dry critical assembly tests will be used to support the design of the prototype module and verify the reactivity feedbacks prior to the safety tests. Sodium in critical tests will be completed in the prototype module prior to conduct of power operation.

1.4 Comment

Please explain why the following regulatory guides are not included in Table 1.8-1:

1.27	1.106	4.1	5.7	7.1	8.2
1.40	1.125	4.2	5.20	7.4	8.3
1.52	1.136	4.7	5.29	7.6	8.4
1.178	1.151	4.8	5.43	7.7	8.6
1.95	1.152	4.11	5.44	7.8	8.8
1.98	1.153	4.15	5.54	7.9	8.10
1.99		5.1	5.62	7.10	8.19
					8.29

Response

Table 1.8-1 was reviewed to check its completeness against the listed 42 Regulatory Guides. Regulatory Guides 1.151, 1.152, 1.153, 4.2, and 4.7 were added to Table 1.8-1. Regulatory Guides 8.2, 8.6, 8.8, and 8.12 were removed from Table 1.8-1 on the basis that they apply to the owner/operator of the plant and do not have major impact on the design. We agree that most of the 4.0, 5.0, 7.0 and 8.0 series Regulatory Guides listed would be applicable to the owner operator of a PRISM plant, but they have not been added for the same reason that the Division 8 series regulatory guides were revisited. An ammended Table 1.8-1 has been submitted with Amendment 2.

The applicability of the following three Regulatory Guides have been increased as discussed.

1.98 ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFF-GAS SYSTEM IN A BWR

PRISM does employ an off-gas system. The gaseous rad waste associated with PRISM will be small compared to a BWR; however, the intent of this guide is being followed in terms of radiological evaluations associated with failures in the cover gas cleanup system.

1.106 THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR OPERATED VALVES

Although PRISM does not require the operation of safety related valves during operation or upon shutdown (it operates in a hermetically sealed state) it does employ normally closed remotely actuated isolation valves on both the cover gas the primary sodium cleanup lines.

1.125 PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR PLANTS

The PRISM R&D program includes several thermal hydraulic key feature tests and the results of those programs will be available in preoperational licensing discussions. However, since the PRISM prototype reactor will be operated as a safety test prior to certification, the thermal hydraulic performance will be fully understood.

1.5 Comment

There is no mention of the NRCs standard review plan (SRP) in the PSID. The Commission's Advanced Reactor Policy requests that existing criteria and methodologies be utilized, where practical. The SRPs have long been an important part of the staff review for LWRs. It was used extensively in the CRBR review. Further, 10 CFR 50.33 (g) directs us to examine the capability of a design to meet the acceptance criteria identified in the various SRPs. We believe you should be prepared to address the applicability of the SRPs on a case by case basis as we progress through the review of your PSID. Deviations from SRP practices should be explained. These results should be documented in an appropriate way in the PSID.

Response

During the preparation of the PRISM PSID, it was assumed that the NRC would use the SRP as guidance in its review of the document. There was no need to note this assumption and since the PSID level of design detail and the level of review was not expected to focus on SRP acceptance criteria, these have not played a major role in preparation of the PRISM design or the PSID. Application of requirements are discussed in Principal Design Criteria, Section 1.2.1; NRC Regulatory Guides, Section 1.8; NRC General Design Criteria, Section 3.1 and Classification of Structures, Components, and Systems, Section 3.2. SRP requirements can be addressed on a case by case basis but we are not prepared to commit to the SRP level of requirements at this stage of design.

1.6 Comment

In Sections 1.1 and 1.2 of the PSID you refer to "design certification of a standard PRISM design" by the NRC. This topic will be one of the subjects for discussion at a meeting later this year on Chapter 14 (Safety Test Program). In anticipation of that meeting it would be helpful to identify in Chapter 1 those specific portions of the plant for which you intend to request Design Certification.

Response

The subject of scope and extent of the standard plant design for which design certification will be requested has been the subject of discussions with both the NRC and the staff. The basis for the PRISM design is to standardize the design and factory fabricate to the maximum extent practical. Thus, the conceptual design is being developed with the objective of standardizing the design in all areas that do not have site specific interface requirements. Text and figures have been added to section 1.9 to identify specifically the current plans for certification scope and extent.

2

2.1 Comment

Since you intend to minimize the role of evacuation for your plant you should include the PAG dose limits to the relevant subsection of 2.2.3.

Response

Same as Response to Comment 1.2 on page F1-1.

3

3.1 Comment

We note that some of the criteria are not identical to the corresponding LWR GDCs in Appendix A of 10 CFR 50. For example, Criterion 21 is quite different and appears to be written specifically for the PRISM design. Please explain the basis for these changes and identify any other criteria that have been modified. Such changes may be consistent with our recommendation regarding the development of PDCs for PRISM (see question 1.1) and your response to this question may be combined with your response to question 1.1.

Response

Typos exist in Criterions 19, 20, 21, 28 and 55. They will be corrected in Amendment 2 as follows:

Criterion 19, page 3.1-18, "all postulated" will be eliminated in the first paragraph. The word "all" in the last paragraph will be eliminated.

Criterion 20, page 3.1-19, in the title will read "Systems".

Criterion 21, page 3.1-21, two sentences are missing in the beginning of the first paragraph starting with "The protection system shall be designed for high functional reliability...," and ending with "protection system can be otherwise demonstrated."

Criterion 28, page 3.1-28, in item "2.", portion starting with "These postulated reactivity...," will be made a separate paragraph.

Criterion 55, page 3.1-47, in item "4.", parenthesis will be removed.

3.2 Comment

For the reasons noted in items 1.2 and 2.1 above you should include the PAG dose guidelines to the considerations of containment performance evaluation in Criterion 16 on page 3.1-13..

Response

Same as Response to Comment 1.2 on page F1-1.

3.3 Comment

The evaluation against Criterion 19 given on pages 3.1-18 and 3.1-19 imply that an operator is not required. We believe this should be clarified to indicate the important role of the operator even in a highly automated and inherently safe facility. While the precise functions of the operator at such a facility will evolve through reviews (such as the one we are now pursuing), and ultimately through operating experience, it is likely that, as a minimum, the operator will provide timely assessment of data and communication with authorities during abnormal operation, provide a backup to the safety systems and initiate recovery action following an abnormal event. We note also our recommendation that the remote shutdown facility be safety grade and include appropriate power, control, instrumentation, lighting, environmental and communication facilities. You should be prepared to discuss this subject further at our meeting on PSID Chapter 7.

Response

There was no intention in the evaluation of Criterion 19 to imply that operators are not required at the site. The operator's role in the highly automated and inherently safe PRISM plant is still important and extends to the functions of timely assessment of data and communication with authorities during abnormal operation. Operators are also available to provide a backup to the safety systems and certainly operational staff will be required for recovery from an abnormal event. The exact role and tasks of the operators at the site as well as the operating staff that is not onsite can only be determined with further detailed analyses. The evaluation to Criterion 19 has been revised with the intent to more accurately communicate the changing but still important role of operators in the PRISM plant, during both normal and abnormal occurrences.

The inherently safe response of the reactor and the reduction in potential consequence of severe accidents and the level of automation being designed into the plant are expected to be particularly important in resulting in a change in the roles of operators.

The indefinitely long grace periods that are associated with even severe accidents allow the diagnostic and recovery tasks to potentially be assigned to staff other than main control room operators. The role of the main control room operator in the abnormal events is expected to be both reduced (shared with other knowledgeable staff and simplified) and spread out in time so that operator overload will not occur.

3.4 Comment

In the evaluation against Criterion 20 on page 3.1-20 you note that the coastdown of the primary pumps is required on every reactor trip to protect against thermal transients to assure a satisfactory operating life. What is the role of pump coastdown with regard to fuel performance in the event of a loss of heat sink or station blackout event.

Response

Under scram conditions it is desirable to turn the pumps off to avoid a thermal shock to components. Failure to turn the pumps off following any scram that would continue to supply full electric power to the pumps would produce the life reducing thermal shock. Station blackout would not be one of these transients since the pump power would not be available. Failure to trip the primary pumps for loss of heat sink cannot be tolerated for an indefinite time without causing damage to the pumps because in addition to causing the initial down transient shock, they add considerable heat to the system. This is not a problem to the fuel.

The evaluation of Criterion 20 refers to thermal shock of the reactor structural components with the reference to the RPS flow coastdown that accompanies the scram. The thermal shock can be life limiting to the fuel but it has not been evaluated yet. The principal core component susceptible to this damage mechanism during a scram is the handling socket. More detailed anlaysis will be required to determine whether the thermal transient during a scram without a concurrent flow coastdown will yield acceptable handling socket damage or whether the flow coastdown will be required by the core components as well as the reactor structures.

3.5 Comment

We believe the PRISM design should at least meet the intent of GDC-44 (page 3.1-41). While water is not used as a means to transfer heat or as an ultimate heat sink clearly the function is important. It is probably addressed by the RVACS. This criterion complements the residual heat removal criterion number 34.

Response

The evaluation against criterion 44 in Section 3.1.4.15 has been changed with Amendment 2 of the PSID. We will be meeting the intent of GDC 44 with RVACS but we see no value in keeping GDC 44 in the PRISM PDC'S. The complimentary role it has to GDC 34 has been included in the equivalent PRISM PDC.

3.6 Comment

We note that, while you have identified those structures, systems, and components in Table 3.2-1 considered to be nuclear safety-related, this table does not identify the Quality Group (Safety Class), Seismic Category, Construction Code, or Quality Assurance applicable to these items. There is, therefore, little to review at this stage of the presentation and we are therefore unable to judge the adequacy of the safety-related structures, systems and components. To help us in our review please provide in Section 3.2 a table summarizing the safety and seismic classifications of the various structures, systems and components including the appropriate codes and standards (ASME, IEEE, etc.) that are applicable.

Response

The following classifications developed from the CRBRP are added to Section 3.2 and noted in Table 3.2-1 Safety-Related Equipment and Structures.

Safety Class 1

Safety Class 1 (SC-1) applies to those components that:

1. Comprise part of the primary coolant boundary;
2. Are used to perform scram functions under any plant conditions;
3. Maintain core geometry or provide core support and whose failure could initiate a core disruptive accident.

Safety Class 2

Safety Class 2 (SC-2) applies to any component not in SC-1:

1. That is required to maintain an adequate reactor coolant inventory following a primary coolant boundary leak;
2. That is a part or extension of the reactor containment boundary;
3. That is required to remove residual heat from the reactor core,
   - whose single failure following any plant condition constitutes a loss-of-safety function, or
   - that is not normally operating or cannot be tested adequately during normal power operation.
4. The single failure of which could cause a loss-of-safety function of other SC-2 components.

Safety Class 3

Safety Class 3 applies to those components not in SC-1 or SC-2:

1. That are required to remove residual heat from the reactor core;
2. The failure of which could result in the loss of safety function of another component (e.g., loss of cooling to components which require cooling for accomplishment of their safety function);
3. That are extensions of the primary coolant boundary and are capable of being isolated from that boundary during all modes

of normal reactor operation by two valves, each of which is either normally closed or capable of remote closure;

4. The failure of which could result in the release to the environment of radioactivity and would result in potential off-site exposures that are comparable to the guideline exposure of 10CFR100.

The requirements of 10CFR50 Section 50.55a and Regulatory Guide 1.26 define a correspondence of Quality Groups (QG) and ASME Code Classification for piping and components such that QG-A corresponds to ASME Code Section III - Class 1, QG-B corresponds to Section III - Class 2, QG-C corresponds to Section III - Class 3. Interpretation of these guidelines is necessary for certain systems and components by taking cognizance of the differences in plant design and technology between the LMRs and LWRs. In these cases, conservative interpretations of the intent of Regulatory Guide 1.26 is made. Quality Assurance requirements of 10CFR50 Appendix B are described in Chapter 17.

Seismic Category I

Safety-related structures, systems and components are designated Seismic Category I in conformance with the applicability of Regulatory Guide 1.29.

3.7 Comment

Why are fuel storage instrumentation and control systems and electrical systems not listed in Table 3.2-1?

Response

Fuel Storage

Ex-vessel fuel storage, such as the fuel handling cell, uses natural ambient cooling to maintain the fuel and blanket assemblies within desired temperature limits. Since the building structures are designed as seismic Category I there are no accidents identified that could result in off-site exposures comparable to 10CFR100 limits. The fuel storage equipment and structures are therefore currently classified as non-safety related but seismic Category I.

Instrumentation and Control Systems

The majority of safety-related instrumentation is provided by the Reactor Protection System. These items were included in Table 3.2-1 under the Reactor Protection System. Similarly, the majority of safety-related control actions are executed by the Reactor Protection System. Its multiplexing hardware and control logic is contained in the cabinets listed under the Reactor Protection System in Table 3.2-1.

Electrical Systems

Safety-related electrical power equipment is described in Chapter 8.0, Sections 8.3.1 and 8.3.2 and EM Pump coastdown equipment in Section 8.3.3. These systems are listed below, and have been added to Table 3.2-1 with Amendment 2.

VII. ELECTRICAL POWER

Class 1E DC Subsystem
Class 1E AC Subsystem
Electro-magnetic Pump Power Supply

3.8 Comment

In Section 3.9.1 you refer to Appendix D for the design transients used to evaluate the mechanical systems and components discussed in Section 3.9. Please provide the duty cycle for replaceable components. Also provide a discussion of how the plant is designed, in this connection, for ATWS and station blackout events.

Response

The Electromagnetic (EM) Pump has a predicted life of 30 years while the Control Rod Driveline has a life of 20 years. The duty cycles for an EM Pump are 1/2 the frequencies for the events stated in Appendix D. The Control Rod Driveline duty cycles are 1/3 the frequencies stated in Appendix D. Fractional frequency values are rounded to the next whole number.

All ATWS events are bounded by the beyond design basis events (BDBE) described and evaluated in appendix E of the PSID. The events are:

- Unprotected loss of primary flow and loss of IHTS cooling (ULOF)
- Unprotected loss of IHTS cooling (ULOHS)
- Unprotected control rod withdrawal (UTOP)

Based on their very low probability of occurrence, those events are considered beyond the design basis and are included in the design process to assure public safety.

As discussed in Appendix E of the PSID, the criteria used to judge the adequacy of the reactor performance for these BDBEs are based on providing for public safety by assuring the integrity of the fuel rods and the primary system structures. The criteria consider the duration of two key periods during the accident transients. For some transients there is a brief interval shortly after the start of the event during which the highest temperatures occur. For the brief highest temperature period of the transient, the most likely cladding midwall failure mechanism is expected to be stress-rupture due to weakening of the HT-9 cladding at high temperature. For this situation, the cladding midwall temperature limit is 1450°F. For the longer period of the transient at lower temperatures, the most likely cladding failure mechanism is the formation of a low-melting point eutectic between the cladding and the metal fuel. The fuel-cladding interface temperature limit for this situation is 1290°F.

Accommodation of a blackout event is discussed in Appendix A of the PSID (Initiating Event 16: Station Blackout). As stated in Appendix A, the station blackout event is defined as loss of the capability to provide electric power sufficient to remove the operating power heat load. This means loss of all off-site and on-site electric power sources capable of running the BOP, IHTS, and primary pumps. PRISM is designed so that its normal response to loss of all off-site power sources is not to shutdown to decay heat levels, but simply to throttle back to 10% power and run with on-site power.

3.9 Comment

Please explain the application of the leak-before-break criteria with regard to a) public safety and b) plant economics.

Response

a) Public Safety

The reactor vessel is designed, constructed and code stamped in accordance with the ASME, B&PV Code Section III, Subsection NB. The reactor vessel is designed to safely accommodate steady state and transient conditions. The reactor vessel will also be designed to withstand the OBE seismic condition as a service level B event. Material selection requires consideration of compatibility with the coolant fluid and any degradation of material properties during service.

The material selected for the reactor vessel is stainless steel 316 and is expected to behave in a non-brittle manner for the conditions defined, and also postulated accident conditions, with substantial margins. Process control and materials characterization as specified in Reg. Guide 1.44 shall prevent the possibility of intergranular stress corrosion cracking (ICSCC) by eliminating the use of sensitized (including weld sensitized) material during purchase of materials, fabrication and installation of the vessel. Cracking by this mechanism is not a potential problem.

Both analytical and experimental work has demonstrated that if a crack propagates in stainless steel it will tend to penetrate the wall with minimal extension. The final crack length at time leakage is detected will be small relative to the critical crack length, i.e., the length at which rapid enlargement could occur. Therefore, a leak in the reactor vessel will be detected when crack size is substantially less than the critical crack size.

Inservice inspection will be governed by the ASME Code, Section XI, Division 3. The primary method of examination on the reactor vessel will be by Continuous Monitoring utilizing sodium ionization detection, spark plugs and pressure sensors. These methods will provide for early detection and warning with sufficient time for the operator to take corrective action. Upon detection of a leak into the annulus, plugging filters will be analyzed to verify the indication. Monitoring information of the reactor cover gas and sodium level will also be available for diagnostic evaluation along with pressure monitoring data on the containment vessel. This method of inspection will assure the integrity of the component is maintained throughout the life.

b) Plant Economics

Since the IHTS and SGS are not nuclear safety related, fracture prevention of the intermediate sodium coolant and steam/water pressure boundary is for normal plant safety and economics. The IHTS and SGS piping and components are designed, constructed and code stamped in accordance with the ASME, B&PV Code, Section VIII and B31.1. The design is required to meet the steady state and transient structural requirements imposed on it by the reactor system. In addition, the systems will be designed to withstand the DBE seismic loads. In addition, the IHTS is designed to withstand full steam pressure at design temperature under faulted conditions. Ductile material have been selected; austenitic stainless steel for the IHTS and 2 1/4 Cr-1M for the SGS. Therefore, the systems are designed to behave in a non-brittle manner under all conditions up to and including postulated accident conditions with suitable margin as provided by the Code.

The leak-before-break characteristics of these ductile materials allow detection in the initial stages of a leak. This is an important factor to provide a warning and sufficient time for operator corrective action to mitigate the enlargement of the leak and the severity of the coolant spill. The IHTS will be equipped with liquid metal leak detectors which will provide indication and location information for liquid metal-to-gas leaks. Contact detector and plugging filter detectors will be provided to monitor the space around the piping, valves and major components. The SGS will be equipped with redundant hydrogen detectors to provide indication of steam/water-to-sodium leaks. In addition, smoke, temperature and pressure detectors located within the equipment vaults will provide data to assist in verifying the existence of a leak and establishing its location.

The in-service-inspection requirements for liquid metal service will conform to the ASME Code, Section XI, Division 3, Class 3 and for steam/water service will conform to the ASME Code, Section XI, Division I, Class 3. These requirements will assure that structural integrity is maintained throughout the operating life of the systems.

3.10 Comment

Referring to Section 3.5.3 in connection with the overall damage prediction of missile barriers subjected to impactive or impulsive loads, do you consider non-linear response of structural elements (reinforced concrete or structural steel)? If yes, we suggest you include a commitment in PSID that assumption of non-linear response will meet or exceed the criteria of permissible ductility ratio in Appendix A of SRP Section 3.5.3.

Response

Non-linear response of structural elements is considered for overall damage prediction of missile barriers subjected to impactive or impulsive loads in accordance with Reference 3.5-3 of PSID. The last sentence of Section 3.5.3 will be changed in Amendment 2 of the PSID to "The design procedures are in accordance with Reference 3.5-3, except that the non-linear response satisfies the criteria of permissible ductility ratios designated in SRP Section 3.5.3, Appendix A."

3.11 Comment

In 6th line of Section 3.7.1.5 there is a typographical error. The word "Section 3.7.II.1.b" should be "Section 3.7.1.II.1.b."

Response

The word "Section 3.7.II.1.b" will be revised to "Section 3.7.1.II.1.b" in this amendment to the PSID.

3.12 Comment

For the soil-structure interaction (SSI) analysis methods in Section 3.7.2.1 you have indicated the use of SASSI computer program. This method is formulated in frequency domain using the complex frequency response method. The staff would like to know the details of the assumptions and technical approach used in the SASSI computer code. The acceptance of this method will be based upon the review of details, demonstration of proper implementation of this method, and adequate parametric and sensitivity studies. Moreover the staff would like to mention for your benefit that the SRP Section 3.7.2 is being revised after the SASSI Workshop held in June 1986 sponsored by NRC. The proposed revision will be sent out for public comments in the near future. This proposed revision has alternative approaches for SSI methods acceptable to the staff.

Response

SASSI is a three-dimensional soil-structure interaction computer program using finite element models and complex frequency response methods. Documentation of PRISM applications of this program during detailed design will include a detailed description of the technical approach and assumptions used and a demonstration of the proper implementation and application of the methods of the program. Results of parametric and sensitivity studies performed to identify significant response parameters and to address uncertainties, as well as to give response insight, will be provided. Additional parametric studies will be performed to test program capabilities, if needed. Other methods for evaluating SSI, acceptable to the staff, will be considered.

3.13 Comment

In the third paragraph of Section 3.7.2.2 peak broadening of $\pm$10% is mentioned. SRP Section 3.7.2 and Regulatory Guide 1.122 both require peak broadening of $\pm$15%. What are your justifications for non-conformance of SRP requirements? Without acceptable justification $\pm$15% should be used.

Response

SRP Section 3.7.2.II.9 states that any reasonable method for determining the amount of peak widening associated with the structural frequency can be used, but in no case should the amount of peak widening be less than $\pm$10%. For PRISM, seismic floor response spectra analysis considers significant variations in soil properties (shear wave velocities, shear modulus, damping and density) including soil structure interactions, which represent the main contributors to frequency variations in response spectra peaks. When a specific site is identified, the required seismologic, geotechnical and geologic data will be used in the development of floor response spectra (FRS). These will be compared with the PRISM envelope FRS to assure the site-specific FRS fall below the envelope FRS (see PSID Section 3.7.1.3). Following this procedure a $\pm$10% peak widening was judged to be adequate for most soils. To increase the applicability of a standard PRISM design, a $\pm$15% peak widening will be adopted, and the last sentence of the third paragraph of Section 3.7.2.2 will be amended to "To variations, the peaks of the spectra obtained from the analysis are widened by $\pm$15%."

3.14 Comment

You make no mention in Section 3.7.2.4 of how accidental torsion is accounted for in the seismic design. SRP Section 3.7.2, Subsection II.11 requires that an additional eccentricity of 5% of the maximum building dimension shall be assumed to account for accidental torsion.

Response

Section 3.7.2.4 will be amended as follows to comply with SRP Section 3.7.2.II.11:

- o Delete first paragraph

- o Revise second paragraph to "In the analysis of structures, torsional effects are incorporated into the dynamic model. This includes an additional seismicity of ±5% of the maximum building dimension to account for accidental torsion."

3.15 Comment

Line 3 of Section 3.8.2.2 states "Consistent with Reference 3.8-1, earthquake loads are not combined with loads resulting from a postulated pipe break." Reference 3.8-1 in PSID is NUREG-1061 which does not state that earthquake loads may be decoupled from pipe break loads for design of structures. With this consideration, your table 3.8-1 should be revised for load combinations 12 and 13 to include OBE and SSE loads. Similarly, Table 3.8-2 should be revised for load combination 10 to include OBE loads, and another load combination 11 should be added to include pipe break loads plus SSE loads. This requirement is consistent with SRP Section 3.8.4 and is unaffected by the guidelines of NUREG-1061 which relax the load combination criteria for piping design only.

Response

The simultaneous occurrence of a pipe break and either a SSE or OBE was considered too improbable to be incorporated in the design. Moreover, it is intended to apply leak-before-break technology to both moderate and high-energy piping systems as discussed in PSID Section 3.6.2.1. Given this consideration, the impact of simultaneous treatment of pipe break and earthquake forces is inconsequential. However, the PSID will be amended in the next revision as follows to comply with SRP Section 3.8.4 load combination criteria, and be available for application to those pipine systems which fail to satisfy the leak-before-break criteria:

- o Section 3.6 - Delete second paragraph, Page 3.6-1 "The simultaneous occurrence ... in Section 3.8."

- o Section 3.8.2.2, Reinforced Concrete structures - Delete third line "Consistent with Reference 3.8-1 ... postulated pipe breaks."

- o Section 3.8.2.2, Structural Steel Structures - Revise fifth and sixth lines to "As stated above for concrete structures, pipe break reaction loads are expected to be zero."

- Table 3.8-1 - Revise load combinations 12) and 13) to read:

  12) $U = D + L + T_a + R_a + 1.25P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E_o$

  13) $U = D + L + T_a + R_a + P_a + 1.0\ (Y_r + Y_j + Y_m) + E_s$

- Table 3.8-2 - Revise load combination 10), and add load combination 11), as follows:

  10) $1.6S = D + L + T_a + R_a + P_a + 1.0\ (Y_r + Y_j + Y_m) + E_o$

  11) $1.7S = D + L + T_a + R_a + P_a + 1.0\ (Y_r + Y_j + Y_m) + E_s$

<u>3.16 Comment</u>

Referring to Section 3.8.4, what are your design considerations for spent fuel pool racks? Nothing is mentioned in PSID. It would seem appropriate that the PRISM spent fuel storage meet the criteria given in Appendix D to SRP Section 3.8.4, unless justification can be provided otherwise.

<u>Response</u>

In the PRISM reference plant, spent fuel is transported directly from the reactor to an <u>on-site</u> fuel reprocessing facility. Hence, the criteria given in Appendix D to SRP Section 3.8.4 were not incorporated into the PSID. However, there is now consideration of an option of <u>off-site</u> fuel reprocessing facilities, in which case spent fuel will be stored in an inerted, natural circulation cooled cell in the reactor service building prior to shipment. To be prepared should this option be invoked, the PSID will be amended to comply with the intent of Appendix D to SRP Section 3.8.4 insofar as these provisions apply to PRISM spent metal fuel storage racks.

<u>3.17 Comment</u>

Referring to Section 3.8.5 please describe the foundation design of Seismic Class I Structures, what bearing capacity have you used for sizing the foundations? If some specific number in pounds per square foot is used, why is it not mentioned in PSID. If no specific number is used, how do you design the foundations? How will you verify that the bearing capacity of a site is adequate for a specific building foundation?

<u>Response</u>

Allowable foundation bearing values are based on the hypothetical Middletown site. Once a specific site has been selected, soil investigations will be performed to verify the adequacy of the foundation bearing values used in the design. Section 3.8.5 will be amended following paragraph:

"Foundation design is based on the hypothetical Middletown site. Soil profiles for the site show alluvial soil and rock fill to a depth of eight feet; Brassfield limestone to a depth of 30 feet; blue weathered shale and fossiliferous Richmond limestone to a depth of 50 feet; and bedrock over a depth of 50 feet. Allowable soil bearing is 6,000 psf and rock bearing characteristics are 18,000 psf and 15,000 psf for Brassfield and Richmond strata, respectively. When a specific site is identified, soil investigations will be performed to verify bearing values used in the design."

3.18 Comment

It is not clear to us that the reactivity control and shutdown rod systems satisfy the diversity requirement in Criterion 26. They are both mechanical systems and both depend on rods moving into the core. While it is possible that the inherent shutdown may satisfy this requirement it has not been fully reviewed in this role (as a safety grade reactivity shutdown system) before and it may not operate (or be required to operate) under exactly the same conditions anticipated in the design of control and shutdown rod systems. Please provide additional justification for your evaluation. Your response can be in general terms since detailed reviews will be made in connection with Chapter 4, 7, 15 and Appendix A and E.

Response

The PRISM reactor is provided with six control rods, each with sufficient worth to permit shutdown from hot full-power condition to cold shutdown condition, even if the remaining five rods were withdrawn to the normal full power position. The rods are operated by two independent and diverse systems: the plant control system (PCS) and the reactor protection system (RPS). As discussed in Section 7 of the PSID, the PCS and the RPS are provided, in turn, with redundant and diverse components to perform their functions (sensors, computers, software logic, scram circuit breakers, electric power supplies).

The PCS provides the capability for power shimming and load following by moving one control rod at a time. It also provides the capability for power rollback to cold shutdown or hot standby condition by inserting all the rods into the core using two redundant control motors. The RPS provides backup shutdown capability by scramming all rods into the core. Shutdown can also be realized by manual scram from the RPS vaults, the control center, and the remote shutdown facility (see Section 7).

As discussed in p. 3.1-26 of the PSID, diversity of the scram function is provided by two scram methods for each control rod. The first scram method is by release of an electromagnetically supported scram latch that decouples the absorber bundle from the driveline within the control assembly. Electric power to hold the scram latch closed is provided by normal site power. Should the reactor protection system interrupt the power or should site power be lost, the latch will open and the absorber bundle will fall into the core by gravity. The alternate shutdown method is by an electric motor within the drive mechanism that drives the absorber assemblies into the core. The motor power is supplied by an independent Class 1E source and has sufficient strength to overcome the control shim drive motors and to exert a high force on the absorber assembly (see Section 4).

Use of the six control rod units for both control and shutdown enhances the shutdown reliability as a result of:

1) the frequent testing of the control rod motion offered by the rods use for power shimming and load following.

2) the availability of three drive rod insertion mechanisms (control motors, scram drive-in motor, and magnetic latch) for each rod.

As shown in Appendix A, these features have resulted in:

1) an extremely reliable control rod unit, and

2) an extremely reliable shutdown capability which has access to six control units although one unit is adequate for shutdown.

It is true that in addition to the redundancy and diversity provided in the shutdown systems there is the additional inherent shutdown capability of the reactor. The performance of this charcateristic is judged with different criteria than those used for the mechanical systems.

<u>3.19 Comment</u>

While the RVACS appears to be an effective decay heat removal system, it may not, by itself, satisfy the redundancy requirement of Criterion 34. The ACS is not safety grade and is not referred to in the evaluation of criterion 34 (see pages 3.1-34 and 3.1-35). Please provide additional justification for your evaluation. Your response can be in general terms since detailed reviews will be made in connection with Chapter 5 of the PSID.

<u>Response</u>

It is GE's position that Criterion 34 is satisfied for the PRISM plant. Criterion 34 refers to a safety grade shutdown heat removal system with appropriate redundancies. The RVACS meets these requirements as elaborated on in the following.

First, the RVACS has a failure probability that is several orders of magnitude lower than conventional decay heat removal systems. This low failure probability is achieved by the use of passive natural convection heat removal. Redundancy in RVACS applies only to the flow paths (sodium and air) since there are no active systems involved. RVACS does that by having 1) redundancy in air inlets (4 total), 2) redundancy in air outlets (4 total), 3) redundancy in air flow path around collector cylinder (not currently shown in design drawings), and 4) redundancy in sodium flow paths within the reactor vessel (either through IHX's or through the overflow path between reactor vessel and its liner). In addition, the RVACS will perform its intended function with a significant air flow path blockage and no electrical power is needed to operate RVACS.

The normal heat removal path through the steam condenser for an individual reactor module also has a very low failure probability. The reason for this is the redundancies in the steam supply system resulting from operating three reactor modules in parallel on a single turbine-generator set. When one reactor module is shutdown for some reason, feedwater and other steam equipment loads are cut back in PRISM whereas in conventional plants startup of auxiliary standby equipment is required. The process of cutting back on e.g., feedwater equipment, is inherently more reliable than startup of standby equipment resulting in a reliability advantage for PRISM type plant in the normal heat removal path.

There is also the highly reliable non-safety grade ACS. In this system heat is removed from the steam generator by natural convecting air as in RVACS and heat transport from the reactor to the steam generator is possible by natural convection in the primary and intermediate systems although the normal operating procedure during scheduled maintenance outages would be to use pony motors in these systems.

4

RESPONSES TO NRC COMMENTS ON PRISM PSID CHAPTER 4

4.1 Comment

Provide arguments as to why fuel failure propagation is considered highly improbable for metal fuels. In particular, address the loss of bond sodium and cite experiments in support of the response.

Response

There are several potential "pin-to-pin" failure propagation mechanisms which can be initiated by the failure of the first pin. These include heat transfer disruption, blockage formation and mechanical loadings. Most of these mechanisms have been covered in the oxide development program and the results which showed that propagation is highly improbable are fully applicable here (CRBRP PSAR Chapter 15.4). Three mechanisms that are potentially different for metal fuel than for oxide fuel, are considered here. These are fission gas blanketing, loss of bond sodium and flow blockage due to fuel pin dilation and/or fuel extrusion through the cladding breach.

Metallic fuel elements are fabricated with large fuel-to-clad gaps (~75% smeared density) to allow for volumetric expansion of the fuel. However, the large gap exists only at low burnup when the fuel element gas plenum pressure is low. A dominant fraction of the fission gas generated at low burnup is retained within the fuel structure, causing volumetric expansion of the fuel. For volumetric expansion greater than ~50% (gap closure), the porosity is interconnected and thus fission gas is released to the plenum. The potential to gas blanket adjacent pins early in life while the gap is open is absent because of the low fission gas release. Late in life, the metallurgical bonding of the fuel to the cladding or at least no significant residual fuel-clad gap should result in high resistance gas flow and thus obviate the potential for gas blanketing. Late in life, failure in the plenum region (top of pin) would be benign because the inventory of gas is not sufficient to perturb the coolant regime enough to cause propagation.

Sodium and metallic fuel show complete compatibility so that the potential for cladding dilation due to reaction products is absent. Only dilation due to FCMI or creep due to the hydrostatic gas pressure are present. In addition, the porous structure generated in metallic fuel due to the fission gas appears to be structurally weak so that FCMI has been minimal.

Loss of sodium from a fuel pin will degrade performance and increase fuel temperatures but will not result in pin failure for normal operation and design basis transient events. After 3 at % burnup, the fuel is in direct contact with the cladding. The fuel porosity is interconnected and filled with sodium. Loss of sodium would result in gas voids within the fuel and a lower effective fuel conductivity. However, this will not be worse than at about 1.5 at % burnup, where the pores have formed but have not interconnected and filled with sodium, which is used as the basis for all peak steady-state temperature calculations. Prior to 3 at % burnup and fuel-clad gap closure, loss of sodium will result in a gas-filled gap and a significant temperature drop from the fuel surface to the cladding. This is not

4.1 Response (Continued)

calculated to be sufficient to result in fuel melting during normal operation and design basis events.

The potential for fuel extrusion or fragmentation from a cladding failure with a resultant reduction in coolant flow also appears from current data (Reference 1, 3) to be minimal. Metal fuel which operated beyond clad failure in the fuel region for up to 60 days showed no evidence for other than benign behavior. If fuel extrusion should occur, it should be a slow process allowing for corrective action. The results of the recent Camel II C9 test with U-5 wt% Zr fuel support these expectations for metal fuel. No fuel-coolant interactions were observed for a severe TOP-type event in which molten fuel was injected into the tricuspid coolant channels of a seven-pin bundle. Furthermore, although the fuel motion was not dispersive enough to sweep extensive amounts of material out of the pin bundle, the fuel was distributed sufficiently (plated out) into a configuration which allowed uninterrupted coolant flow. Based on the flowrate through the blockage (85% of nominal full flow), the apparent high porosity of the blockage (58%) and the ternary alloy fuel's large thermal conductivity, it is concluded that the debris coolability margin with the metallic fuel planned for use in PRISM appears to be large.

In conclusion, conditions simply do not appear to occur during irradiation that can result in "pin-to-pin" propagation, with metal fuel.

References:
1. R. W. Tilbrook, et al., "Local Fault Tolerance of Metal Fuel," ANL-IFR-37, February 1986).
2. "Status of LMFBR Safety Technology - 1. Fission Gas Release From Fuel Pins," OECD Nuclear Energy Agency, Paris, France, CSNI Report No. 40, February 1980.
3. W. A. Bezella, et al., "Dispersal of Molten Uranium Alloy Under Simulated Transient Overpower Accident Conditions: The Camel II C9 Test," ANL-IFR-67, May 1987.

4.2 Comment

Projected burnups for the metallic fuel with HT9 cladding are much higher than that reached thus far in tests. Please discuss:1) why you feel you can extrapolate from test data using D9 cladding and/or oxide fuels (with HT9 cladding), and 2) plans for testing to higher burnups.

Response

Part 1:

To date oxide fuel with D9 cladding has achieved peak burnups in the range of 15 a/o while oxide fuel clad with HT9 has achieved peak burnups in the range of 13-15 a/o. These burnups approach the range presently being considered for metal fueled reactors - 15-20 a/o. Metal fuel at higher burnups show fission gas release rates comparable to oxide fuel. However, metal fuels at least to current burnup levels

4.2 Response (Continued)

(10 a/o) appear to be structurally less strong than oxide fuel and therefore less able to produce fuel-clad mechanical interaction (FCMI). Current oxide fuel irradiations to 13-15 a/o burnup should be directly applicable to metal fuel with respect to assessing performance since metal fuel produces no more severe and probably less severe conditions on the cladding than oxide fuel.

Part 2:

Current metal fuel testing has shown benign performance at least to current burnup levels; fuel clad with D9 has achieved 10 a/o in EBR-II and fuel clad with HT9 has achieved 5 a/o burnup. However, a question still to be answered for metal fuel is whether at high burnups (beyond 14-15 a/o) solid fission product swelling can cause closure of the interconnected porosity in the fuel. As long as this porosity is open, creep of fuel into the interconnected porosity will prevent the cladding loading from exceeding the plenum pressure. The central void plays much the same role in oxide pins. Once the porosity is no longer interconnected, continued fission product swelling will eventually result in cladding plastic deformation. High burnup run to cladding breach (RTCB) tests currently being run in EBR-II (see Table-Status of Current Metallic Fuel Irradiations in the response to Comment 4 ANL-4) will show whether or not strong FCMI occurs and if it contributes to cladding failure.

4.3 Comment

Do you intend to operate when there are indications of localized fuel failure (fission gasses detected, for example), if so, how much is acceptable and at what failure rate will shutdown be initiated?

Response

When a clad failure occurs, PRISM will continue to operate in run beyond cladding breach (RBCB) mode. Pure gas leakers (generally birth defects) will be allowed to remain in the core until the next refueling outage. Breaches (gas leakers progressing to delayed neutron (DN) emitter) will remain in the core to the next refueling outage or until the cover gas fission gas or primary sodium delayed neutron activity limits are exceeded. Failed fuel will be stored in the reactor for one cycle at spent fuel storage locations.

The fuel for PRISM is expected to be very reliable with a nominal failure rate on the order of 0.2 breaches/reactor/year. Converting this into operating conditions, approximately 85% of the time the reactor will operate without any breaches in the core, 13% of the time with up to one breach and about 2% with up to two breaches in the core. This means that PRISM will operate with failed fuel very infrequently.

Shutdown will be initiated upon indication of a third pin failure or by signals exceeding limits on cover gas activity or delay neutron activity in the primary sodium. These limits will be established in future design phases.

4.4 Comment

Explain your choice of structural materials for the reactor internals and identify which components are of HT9, D9 or SS-316, etc.

Response

The structural material for the core ducts, cladding and wire wraps is HT9. This ferritic stainless steel is chosen for its low irradiation swelling characteristics. D9 is an austinitic steel with greater high temperature creep strength and considerably higher swelling rate. For adequate nuclear and breeding performance in a physically small core, a high fuel volume fraction and long fuel lifetime is necessary. Thus, a low swelling material that permits a compact core is used. The somewhat lower thermal creep resistance of HT9 compared to D9 austenitic steel presents no problem because of the low core operating temperatures and the margin for transient response.

The core support structures and other reactor internals are constructed of austinitic 316 stainless steel. Thermal expansion generated negative reactivity feedback from the core support structure expansion is enhanced by the use of this higher expansion material.

4.5 Comment

Explain why an assembly inlet blockage should be considered incredible. Use detailed drawings of the inlet plenum orificing and the assembly inlet region wherever applicable.

Response

The response to Comment 4.21 discusses the flow blockage prevention features of PRISM in detail. The design features to prevent flow blockage are summarized below.

Figures 4.2-8 and 4.2-9 along with the core map in Figure 4.2-1 show the core inlet plenum and the locations of the core assembly receptacles located within the inlet plenum. The core receptacles consist of a main outer cylinder of two different diameters. The lower part of the larger diameter portion has 6 to 8, approximately 2-inch diameter flow holes (details not finalized) as indicated in Figure 4.2-8. The assembly orificing for those assemblies where this is required is located within the large diameter portion of the receptacles as indicated in Figure 4.2-8. Any debris postulated to entering the inlet plenum through the eight 12-inch diameter feeder pipes will have to find its way to the receptacles and block thoroughly; 1) all the 6 to 8 inlet flow holes, 2) all the orifice holes where such orificing exist (low power assemblies), or 3) be small enough to block all core assembly flow area between fuel pins. As discussed in 4.21 initial sodium fill and filtering will assure that no such debris exists. Because of the remote operations and design to eliminate loose parts, especially those of a size that could block an assembly flow, the potential for reducing flow blockage to incredible low probabilities can be assured. The operation of the

4.5 Response (Continued)

safety test module will confirm the design achievement of the objective.

4.6 Comment

Show diagrams of all core restraints (load route, etc.) and explain how their respective expansions affect the core reactivity, including curves of predicted reactivity change.

Response

The core restraint system design is discussed in Section 4.2.2.3 of the PSID and Figure 4.2-9 shows the overall restraint features.

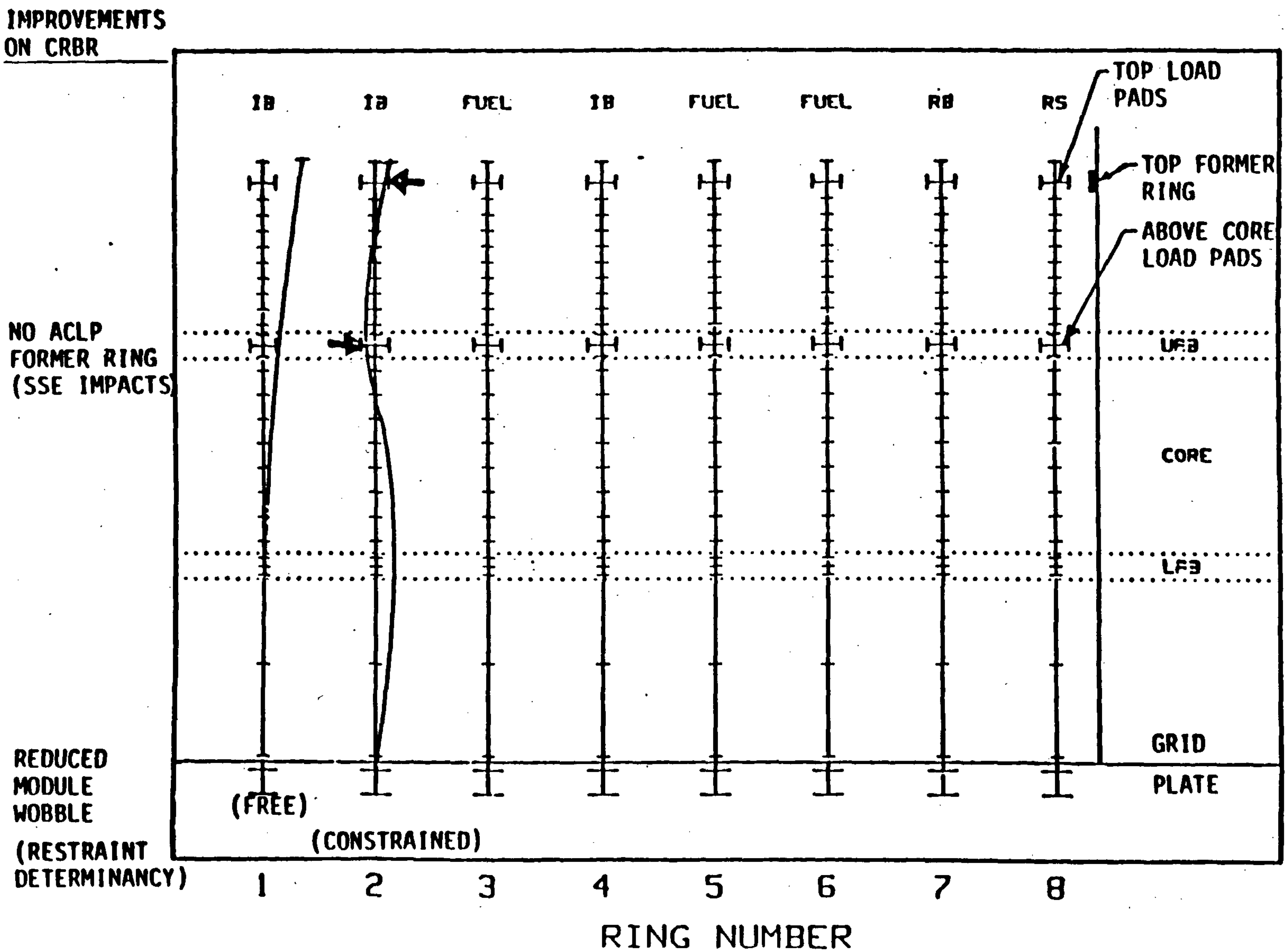
For restraint in the horizontal plane, the assemblies are held by (1) their nosepieces inserted into receptacles in the core support grid plate, and (2) the mutual intersupport of the load pads on the assembly upper ends which are constrained by a former ring attached to the core barrel. The minimum spacing of the assemblies is maintained by a plane of load pads located on the ducts just above the core. The design permits the assemblies to bow in response to the core temperature distribution and provide the bowing reactivity feedback that is one element in inherent reactivity shutdown.

The bowing of the assembly ducts, as constrained by the core restraint system, is illustrated in the attached Figure. The eight rings of assemblies are shown schematically with their top and above core load pads. The top former ring at the elevation of the top load pads is shown; there is no former ring at the above core load pad elevation. Each assembly is secured in a receptacle in the grid plate.

The assembly pictured in ring no. 1 shows the free-bow shape each assembly would take if it were not restrained. It is fixed at the grid plate and "flowers" outward from the core center because the inside face of the assembly duct is hotter than the outside face.

The assembly of ring no. 2 shows the constrained position each assembly is forced into by the PRISM core restraint system. The assembly tries to "flower" outward but is constrained by the top load pads and top former ring to maintain its radial position at the top of the assembly. Core compaction would then result in the region of the active core if it were not for the above core load pads which stop the inward movement at their elevation. The movement caused by the rigid ACLP produces a reverse deflection on the assembly which results in outward bowing in the active core region as the temperatures are increased and, therefore, a negative bowing reactivity feedback. In addition there is an overall expansion at the ACLP plane due to the increased core temperature.

CORE RESTRAINT SYSTEM



4.6 Response (Continued)

The four principal core restraint reactivity feedbacks are (1) bowing of core assemblies due to the ACLP applied moment and expansion, (2) core structural (top former ring) thermal expansion, (3) core support structural radial thermal expansion, and (4) core support structure/control rod driveline axial expansion. These are discussed in the response to Comment 4.28 and in Section 4.6 of the PSID.

4.7 Comment

What is the sodium void reactivity worth, and how was it calculated - perturbation or substitution? Is there any practical means to make it less positive, or preferrably, negative? What is the worst single assembly void worth? Also, if the entire core were voided would there be enough control rod worth to shutdown the reactor?

Response

Sodium void reactivity worth has not yet been calculated by direct Δk methods. Less precise first order perturbation calculations of the sodium void worth produce the following estimates;

- Removal of flowing sodium from all fuel assemblies:
  +3.03$
- Removal of flowing sodium from all blanket assemblies (internal and radial blankets):
  +2.31$

Large LMR cores reduced the sodium void reactivity by reducing the core height to diameter ratio. It is difficult to take advantage of this approach in PRISM for two reasons:

1) High internal conversion for a low burnup reactivity swing requires a neutronically efficient core; i.e., h/d ~1 is desirable.
2) Pancake cores (h/d<1) require a large diametrical space in the reactor dedicated to the core. Such space is at a premium in a compact pool reactor.

These PRISM characteristics contribute to the inherent protection against reactivity accidents and loss of core cooling accidents. We have not found an approach that would retain the inherency advantages in PRISM and also reduce the void worth to a small or negative value.

The void worth of voiding a single assembly (fuel or blanket) is dependent on its position in the core. Calculations for the PRISM metal core have not been made. Based on past oxide core evaluations voiding a single subassembly is estimated to be worth 20-60¢

If the entire core was voided, there would be more than sufficient control rod worth to shutdown the reactor. Full core voiding is worth less than 7$ while the control rod system worth is 22$.

4.8 Comment

Provide control rod insertion (scram) speeds and shape of functions, under either gravitational or motor-driven modes. What is rod position when fully withdrawn? Can bowing affect rod insertion? Describe the absorber rods and provide a rod-worth curve. How were the control and scram worth calculated - by a static or dynamic method?

Response

The control absorber assemblies have not been designed or analyzed in detail. Based on the design and test experience of CRBR and FFTF, it is fully expected that control rods can be designed with acceptable scram performance characteristics. The assumed characteristic is:

Scram drop time ≤ 2 sec

The powered drivein scram time is determined by the drive speed and is currently 2 inches per second with a 36-inch stroke. The bottom end of the absorber bundle is 10% into the active core when the bundle is fully withdrawn. The reference rod-worth curve is provided as Figure F4.8-1. Bowing is not expected to affect rod insertion because large gaps are left between sliding and stationary portions of the assembly. As in CRBRP, detailed alignment criteria will assure that three point contact, which would produce resistance to insertion, will not occur. The control and scram worth are calculated by a static method (see response to Comment 4.13).

The absorber rods are described in Section 4.5.3 of the PSID. There are 61 0.64-inch diameter pins in each assembly. The cladding is 0.06 inch thick and the boron carbide pellets are 0.503 inch diameter. The pellet fabrication density is 92.0% theoretical density.

4.9 Comment

Have any chemical shutdown systems, comparable to boron injection, been considered for your design?

Response

Numerous studies of diverse shutdown systems have been made under the DOE base program and these were reviewed for applicability to PRISM. None were found to be saisfactory from an overall cost and performance standpoint. In addition, as part of a technical trade study on the reactivity control system conducted in 1985, various approaches to self-actuated reactor shutdown were evaluated. These included:

1. A system that injects liquid poison such as Li6 or indium into the coolant in response to overtemperature coolant or high flux.

2. A system that injects solid poison such as B10 granules into the core on overtemperature.

REFERENCE CONTROL WORTH STROKE CURVE

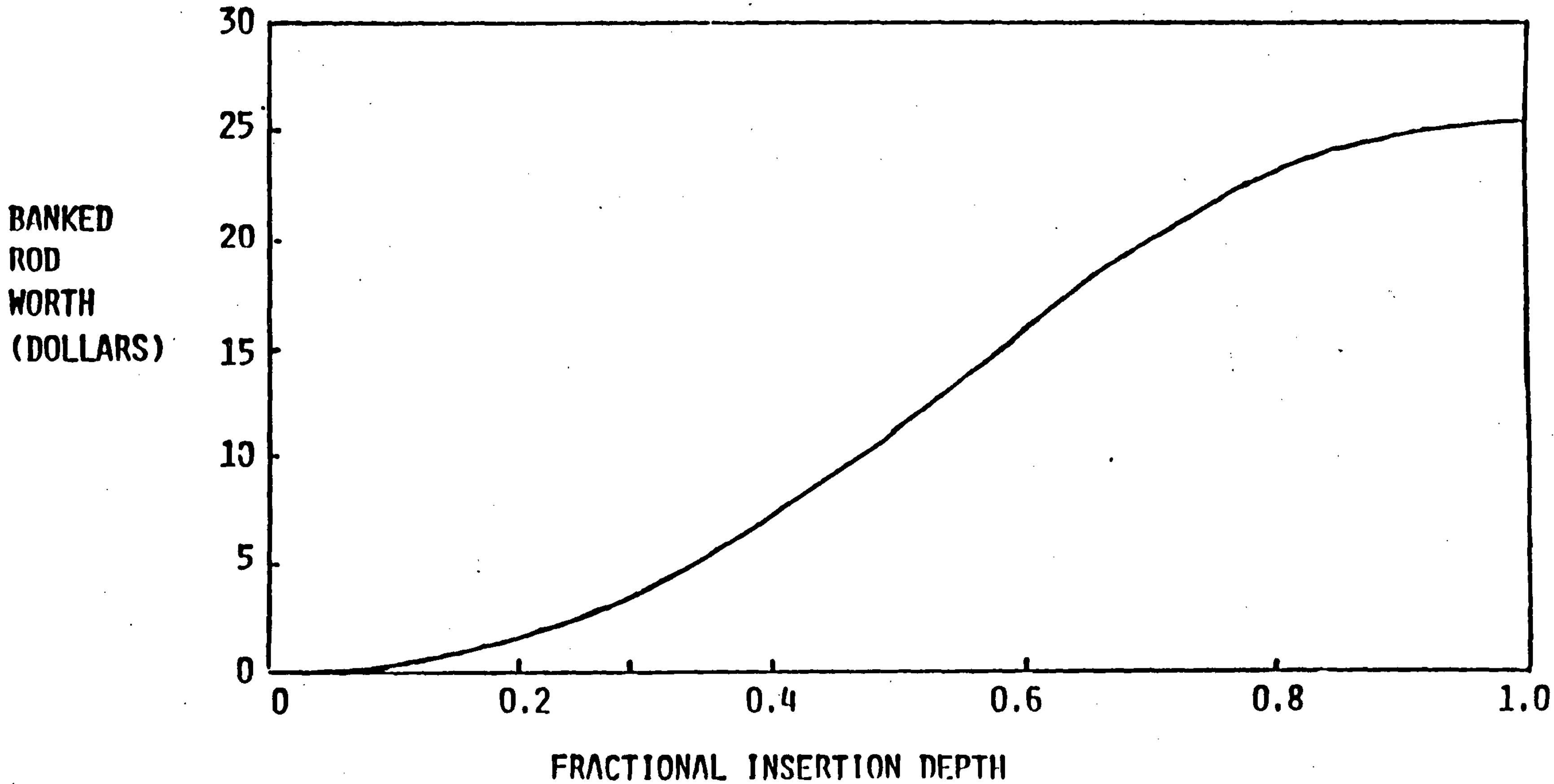


Figure F4.8-1 Reference Control Worth Stroke Curve

87-180-05

4.9 Response (Continued)

3. System that, in response to overtemperature or high flux, dumps liquid poison into closed ducts in an active core region.

4. System which dumps B10 shot, granules or slurry into closed ducts placed in the active core region.

System #1 (liquid poison injection) was rejected because of uncertainty in predicting performance and added cost, complexity and the significant amount of downtime required for poison removal. Until the liquid poison is uniformly mixed with the reactor coolant, there may be rapid changes in reactivity as fluctuating concentrations of poison circulate through the core.

System #2 (poison granule injection) was rejected because the concept is probably a one-shot device which cannot be tested. Cost, complexity and operational considerations discourage its selection.

System #3 (liquid poison injection into closed ducts) was rejected because of testing difficulties. Testing would involve overheating (irradiating) injection trigger during reactor shutdown or would involve overheating (overpowering) entire reactor; the first approach costs time and tooling, while the second involves potentially detrimental effects on reactor and plant. Since the poison is a liquid, there was a concern that it could leak out if the duct was breached.

System #4 (B10 granules into closed duct) was rejected for the same reason as system #3 above. In addition, since the poison is in the form of small particles, there was concern that these particles may compact or bond together during long-term residence in the reactor, such that they would not run freely into the duct when required to do so.

4.10 Comment

After reprocessing, how can isotopic concentrations be guaranteed? What changes in fuel density and contamination can be anticipated? How can this affect the spectrum and the reactivity feedbacks?

Response

Reprocessing will be used to extract plutonium from the blanket and remove fission products from blanket and fuel material. The recovered plutonium will be used to re-enrich the fuel. Reprocessing will not alter the basic isotopic composition of either the uranium or plutonium. The reprocessed fuel fissile content will be controlled by adjusting the plutonium to uranium ratio in the final fuel.

It is not expected that the fuel density will be significantly affected by the small amount of solid fission products present during fuel fabrication. Gas products will be released during reprocessing. The referenced paper evaluates the effect of reprocessing upon core neutronics and should be reviewed for additional details. The

4.10 Response (Continued)

following statement signifies the status of our present understanding of the minimum neutronic effect of reprocessing. "The IFR core static neutronics performance parameters have been shown to be insensitive to variations in pyroprocessing partition and recovery factors which have yet to be established with high precision."

Reference

D. C. Wade and Y. I. Chang, "The Integral Fast Reactor (IFR) Concept: Physics of Operation and Safety," International Topical Meeting on Advances in Reactor Physics Mathematics and Computation, Paris, France, April 27-30, 1987.

4.11 Comment

Have you looked at the effect of fluence on the core barrel? What is the basis for the design limit neutron damage?

Response

The peak total neutron fluence at the core barrel has been calculated to be 1.0×10^{22} n/cm^2. The neutron damage corresponding to this fluence was calculated to be 0.63 dpa based on the neutron energy spectrum and the core barrel material (SS316). Using a value of 2 dpa as the design limit, this indicates a design margin of about a factor of 3 at the end of the 60-year reactor life.

The value of 2 dpa is based on retaining at least 10% RTE in the core barrel at end of life. This design limit corresponds to a fluence limit of 3×10^{22} n/cm^2 based on an average energy of 0.2 Mev for the neutron spectrum incident on the SS316 core barrel. The CRBR SS304 core barrel neutron fluence limit was 1.8×10^{22} n/cm^2 based on an incident neutron spectrum with an average energy of 0.08 Mev. The use of SS316 rather than SS304 in the PRISM core barrel allows accommodation of the harder spectrum and higher fluence limit.

See section 5.3.2.3 of the PSID for a description of the fixed shielding which limits the fluence on the core barrel and other key components.

4.12 Comment

How much core expansion (in inches and/or temperature) can be tolerated before structural failures would be expected? Could the upper grid core support expand faster than the core barrel and crack the barrel?

Response

The top load pads are nominally sized such that, in combination with the gaps between assemblies (and between assemblies and the former

4.12 Response (Continued)

ring), margin to buckling of the duct wall exists for all design basis and selected beyond design basis transients. This margin will be confirmed later as part of detailed design.

The core former ring and core barrel are made of a material (316 stainless steel) with a higher coefficient of thermal expansion than the assembly ducts (HT9). In addition, the gaps between the adjacent ducts and the outer duct row and the former ring are sized to avoid loading the former ring even during BDBA up-transients, where the former ring thermally lags the ducts.

The former ring cannot load the core barrel as they are made of the same material, welded together, and experience the same thermal environment.

4.13 Comment

Describe your core neutronic calculations. How are the cross-sections calculated? What methods are used to calculate K_{eff}, pin powers, burnups? How are reactivity feedbacks evaluated? How is bowing feedback predicted? radial expansion? How are time constants for feedbacks estimated? Can bowing occur in such a manner as to produce positive feedback? What reactivity effects are expected at low power from the free play between the top core load pads and the core former ring?

Response

The methods used in carrying out the nuclear evaluations focus on the cross section generation procedures, the flux solution techniques, burnup calculations, control worth calculations, generation of reactivity feedback coefficients, and decay heat evaluation. This material will be added to the PSID, Section 4.3.3, Analytic Methods and Design Evaluation.

Cross Section Generation

A schematic of the nuclear evaluation process is shown in Figure F4.13-1. The evaluation process is initiated by the generation of regionwise microscopic cross sections utilizing a rapid adjustment technique based upon the self- shielding f-factor approach in the TDOWN data processing code (Ref. F4.13-1). An 80-group preprocessed data library, designated LIB-V(E) (Ref. F4.13-2), was prepared by Los Alamos National Laboratory using the NJOY code system (Ref. F4.13-.3) and the ENDF/B-V data file (Ref. F4.13-4), and was utilized in the TDOWN calculations. The data processing calculations were performed for a heterogeneous cell configuration based upon the multi-region equivalence theory approach in TDOWN. In this approach, the fuel cell was modeled by four regions which included a smeared fuel pellet region, a fuel pin clad region, a sodium region, and a smeared fuel assembly region (i.e., smeared fuel, coolant, cladding and duct material). The 80-group cross sections were homogenized over the cell

4.13 Response (Continued)

regions, and collapsed to few-group cross section files with fluxes generated from several one-dimensional reactor flux solution calculations. Both 6-group and 22-group cross section files were generated with the neutron energy boundaries shown in Table F4.13-1. The reference regionwise temperatures used in the data processing calculations are taken from the core thermal-hydraulic analysis for the steady-state full-power operating condition. Cell calculations were also carried out with elevated temperatures to generate fuel cross sections for calculating the Doppler coefficients. In addition, a set of cross sections was also generated for the sodium void reactivity calculation by removing all sodium from fuel or blanket assemblies.

Flux Solution and Burnup Calculations

All fuel cycle calculations were carried out with the three-dimensional flux solution code DIF3D (Ref. F4.13-5) and the fuel management and burnup code FUMBLE (Ref. F4.13-6). Flux solution calculations were performed using the Hexagonal-Z geometry, and the coarse-mesh nodal diffusion theory approximation to neutron transport. Six energy groups were used for all neutronic computations except for the Doppler and sodium void reactivity calculations in which 22 energy groups were used to minimize the sensitivity of group structure. The fuel cycle computations for the operating interval were performed by a burnup calculation in which the regionwise fluxes and fuel cross sections were taken from converged beginning-of-cycle and end-of-cycle flux solutions and interpolated for several burnup substeps within the cycle. A well converged fuel cycle mass balance solution was obtained by successive iterations of the flux solution and fuel management calculations.

The feed plutonium was assumed to be recycled fuel from Liquid Metal Reactor (LMR). The plutonium and uranium isotopic concentrations are given in Table F4.13-2.

It is noted that in carrying out the burnup calculations, the assumption was made that 95% of the reactor power came from neutron fissions whereas 5% came from capture gammas (based on DIF3D calculations). Thus, burnup data were expressed only in terms of fission power.

A batch-averaged approach was utilized in the iterative process of burnup and flux solution calculations for the equilibrium fuel cycle for fuel assemblies. Discrete shuffling was employed for blanket assemblies. In this approach, it was assumed that each fuel assembly location consisted of several fuel batches, depending upon the fuel lifetime and refueling interval. Thus, the resulting power and flux distributions were interpreted as the average for the equilibrium fuel cycle. The maximum fuel power density can be estimated by utilizing the batch factors, which are defined as the power ratio of fresh fuel to the average fuel. The batch-averaged approach has been found to be fairly accurate in predicting both the peak linear power and the neutron multiplication factors, when compared with the more expensive approach of discrete fuel management calculations.

4.13 Response (Continued)

The coarse-meshed nodal diffusion approach was employed for the Hex-Z flux solution calculations. The validation of DIF3D nodal approximation was carried out by Argonne National Laboratory with the SNR and LCCEWG benchmark problems (Ref. F4.13-7). The results indicated that, for the same geometry model, the coarse- meshed nodal method is comparable or superior to the fine-meshed finite difference method in predicting the k-effective, burnup reactivity, and the power distribution.

In the Hex-Z model, each individual fuel or blanket assembly was divided into 8 axial zones - 4 in the active core section and one each for the upper axial blanket, lower axial blanket, upper axial shield, and lower axial shield. A zone in DIF3D is defined as a set of core regions having the same atom densities. In general, each axial zone contained 2 axial mesh points. Thus, the axial mesh spacing was limited to less than 20 cm, which was considered to be appropriate for accurately predicting the power distribution.

Control Worth Calculation

Control worth calculations are carried out with six neutron energy groups, three-dimensional hexagonal-Z geometry, and nodal approximation in DIF3D flux solutions. The total worths of N (N = 6 for the PRISM cores) control rods can be obtained from the k-effective calculations in a sixty-degree sector of the core layout by taking advantage of the sixty- degree symmetry. On the other hand, the interaction factors which reflect the control rod shadowing or anti-shadowing effect for the cases involving one single rod withdrawn, stuck, or scrammed can only be obtained by carrying out the k-effective calculations in a full core. Table F4.13-3 summarizes the k-effective calculations required to evaluate various control worth characteristics.

Based upon the results of the k-effective calculations defined in Table F4.13-3, the following control characteristics are defined by the identified k_{eff} differences:

W_1 = Burnup reactivity swing = $K_1 - K_3$

W_2 = Total worth of N rods at BOC = $K_3 - K_4$

W_3 = Total worth of N rods at EOC = $K_1 - K_2$

W_4 = Worth of N-1 rods at BOC = $K_3 - K_6$

W_5 = Worth on N-1 rods at EOC = $K_1 - K_7$

W_6 = TOP reactivity (one rod run-out) = $K_8 - K_5$

W_7 = Worth of single rod scram at BOC = $K_5 - K_9$

W_8 = Worth of single rod scram at EOC = $K_1 - K_{10}$.

4.13 Response (Continued)

And, the rod interaction factors at BOEC can be derived as follows:

$$\text{One Stuck Rod} = \frac{W_2 - W_4}{W_2/N}$$

$$\text{TOP} = \frac{W_6}{W_1/N}$$

$$\text{Single Rod Scram} = \frac{W_7}{(W_2 - W_1)/N} \quad .$$

In addition to these calculations, a series of k-effective calculations are performed to generate the control rod worth stroke curve by varying the rod insertion depth. These calculations are needed especially for the designers to determine the feasibility of designing control elements that are shorter than the active core length.

Reactivity Coefficients

Calculations of reactivity feedback coefficients and neutron kinetics parameters are carried out utilizing a series of computer codes. These computer codes include DIF3D, SN2D (Ref. F4.13-8), SNPERT (Ref. F4.13-9) and SNASS, a developmental code at General Electric Company, and are utilized to perform the neutron flux and adjoint solution calculations, perturbation computations, and data manipulations.

Global reactivity coefficients are utilized to normalize the results of mesh-dependent perturbation calculations. The global feedback coefficients are determined by 3-D flux computations using DIF3D Hex-Z geometry and six neutron energy groups. These coefficients include the following:

1. Doppler coefficients (Tdk/dT) are given by:

$$[k(T_2) - k(T_1)] \,/\, \ln(T_2/T_1),$$

where T_2 and T_1 are the temperatures of the cross section sets used in the flux computations. Normally Doppler calculations are performed for various fuel zones to separate the Doppler coefficients for the driver fuel, internal blanket, axial blanket, and radial blanket. As a general practice, global Doppler calculations are carried out using fine group structure (for instance, 22 groups) to minimize the sensitivity of group structure.

2. Density coefficients ($\rho dk/d\rho$) for fuel, structural material, and coolant can be calculated individually in the flux computations by increasing (or reducing) the appropriate isotopic densities by a constant fraction (e.g., 5 or 10 percent). That is,

4.13 Response (Continued)

$$\rho dk/d\rho = \frac{k(\text{perturbed}) - k(\text{reference})}{\text{fraction of change}} .$$

Then the global total density coefficient can be obtained by summing up each individual component.

3. Axial and radial expansion coefficients (Hdk/dH and Rdk/dR, respectively) for constant material mass are computed by uniformly increasing the core size by a constant fraction (e.g., 5 or 10 percent) without changing the material mass. That is, material isotopic densities are reduced accordingly. Then,

$$Hdk/dH = \frac{k(\text{axial expanded}) - k(\text{reference})}{\text{axial expansion fraction}} \text{, and}$$

$$Rdk/dR = \frac{k(\text{radial expanded}) - k(\text{reference})}{\text{radial expansion fraction}} .$$

These two expansion coefficients represent the total effect when expanding the core size, and include two effects of opposite sign -- a positive leakage feedback due to reduction of neutron leakage and a negative density feedback due to reduction of fuel densities. Thus, the leakage components of the expansion coefficients can be derived as follows (using $d\rho/\rho = - dH/H - 2\, dR/R$):

$$(Hdk/dH)_{leakage} = (Hdk/dH)_{total} + (\rho dk/d\rho) \text{, and}$$

$$(Rdk/dR)_{leakage} = (Rdk/dR)_{total} + 2(\rho dk/d\rho).$$

When coupled with the linear thermal expansion coefficients for the fuel, cladding and grid plate, these expansion coefficients can be used to calculate the reactivity feedback effects of core expansion as core temperature changes.

4. Sodium void reactivities can be computed in two alternative ways. The first one is to use the sodium density coefficients from the perturbation calculations and then estimate the effect of voiding the sodium. This approach generally does not accurately predict the sodium void reactivity because: (1) it does not take into account the effect of spectral hardening when the sodium is voided, and (2) the first-order perturbation theory may not be applicable to such a large change in the sodium density. Thus, this approach is only used for a first-order estimation. The second alternative is to perform a direct flux calculation by voiding the sodium while using a sodium-voided cross section set. The flux calculations are usually performed with fine-group (e.g., 22 groups) cross sections to minimize the sensitivity of group structures. This approach is

4.13 Response (Continued)

significantly more accurate than the perturbation method, but the computational cost is also substantially higher.

For the generation of mesh-dependent reactivity coefficients, SN2D is used for the neutron flux and adjoint solution computations. These calculations are carried out in a two-dimensional R-Z geometry. The resulting flux solutions and neutron multiplication factors are then used by SNPERT in the first-order perturbation calculations. The results of these perturbation calculations are the mesh-dependent (R-Z) reactivity data and the neutron kinetics parameters including the prompt neutron generation time, total effective delayed neutron fraction, and the delayed neutron fractions and decay constants for each delayed neutron group. Finally, the computer code SNASS is used to manipulate the mesh-dependent reactivity data and generate the interface data files for the transient analysis.

Decay Heat Evaluation

Decay heat calculations are performed to generate the decay power information for the core transient analysis and for assembly heat loads during refueling movements. The computer code ORIGEN-2 (Ref. F4.13-10) is utilized for the irradiation and decay calculations. A sample calculational procedure is shown in Figure F4.13-2 for a core with a four- year fuel lifetime and a refueling interval of one year.

The decay calculations are initiated by the generation of regionwise total neutron fluxes and the effective one-group neutron cross sections for the uranium and plutonium isotopes. These cross sections are condensed from the results of the multi-group neutronics calculations so that consistent results of power and fuel burnup between the core neutronics analysis and ORIGEN-2 irradiation calculations would be attainable. In general, the reactor core is divided into several regions such as driver fuel (enrichment zone 1, zone 2, etc.), axial blanket, internal blanket, and radial blanket. Then, the irradiation and decay calculations are performed for all regions with their respective loaded mass, flux, cross sections, and fuel lifetime. inally, the total decay power curves can be obtained by summing up the decay power from all contributing regions.

For the fuel region with an n-cycle lifetime, a total of n cycles of irradiation and decay calculations are performed for the charged fresh fuel element (heavy metal and structural material). Each cycle consists of an irradiation period and a shutdown period for refueling and maintenance, consistent with an assumed plant capacity factor. Similar calculations are performed for other core regions with their respective loaded fuel mass, flux level, neutron cross sections, and fuel lifetime. Finally, the total core inventories at BOEC and EOEC are obtained by summing up the results from each burnup cycle and for each core region, and then decay calculations are carried out for a decay time period of up to 5 years to generate the decay heat curves.

4.13 Response (Continued)

Potential Positive Feedback from Bowing

The bowing reactivity feedbacks are discussed in detail in the reponse to Comment 4.28 and will only be summarized here.

Overall, the "linear" power coefficient is negative throughout the power range. Above a power level of about 40% to 60% (may be lower for all but the first fuel loading), when the thermal gradients have become large enough to assure a bound state within the core restraint system, the bowing reactivity feedback is assuredly negative. Until bound up, the core restraint system, theoretically at least, is in an indeterminate state and the potential exists for an assembly to change its direction of lean and create a reactivity change. In addition, starting from a zero power condition, the overall gaps in the restraint planes, combined with the general bow shape, convex to the core centerline, means a slight core compaction must occur during the thermal gradient buildup and, therefore, the restraint system could contribute a small positive feedback during the initial low power portion of the rise to power.

With respect to the potential step reactivity changes due to assembly shift at low power, the maximum amount of positive bowing reactivity feedback occurs at zero power and is approximately +11¢. Even if it is arbitrarily assumed that this amount of reactivity is instantaneously inserted the other feedbacks would inherently counteract and limit the power change to a small amount. The potential step change is small compared to the overall negative power coefficient. As the startup power is increased, the maximum amount of positive bowing reactivity feedback decreases and is nonexistent at 40-60% power and greater.

(4.13 Response - Continued)

TABLE F4.13-1

GROUP STRUCTURE FOR 22 AND 6 ENERGY GROUPS

(a) 22 Groups

Group No.	Upper Energy (eV)
1	2.0000+7
2	3.6788+6
3	1.3534+6
4	8.2085+5
5	4.9787+5
6	3.0197+5
7	1.8316+5
8	1.1109+5
9	6.7380+4
10	4.0868+4
11	2.4788+4
12	1.5034+4
13	9.1188+3
14	5.5308+3
15	3.3546+3
16	2.6126+3
17	2.0347+3
18	1.5846+3
19	1.2341+3
20	7.4852+2
21	4.5400+2
22	2.7536+2
	1.3888-4

(b) 6 Groups

Group No.	Upper Energy (eV)
1	2.0000+7
2	4.9787+5
3	1.1109+5
4	4.0868+4
5	9.1188+3
6	1.5846+3
	1.3888-4

(4.13 Response - Continued)

TABLE F4.13-2

Feed Isotopics (wt %)

Feed Plutonium	Metal Core
Pu-238	0.40
Pu-239	72.44
Pu-240	23.28
Pu-241	2.66
Pu-242	1.22
Feed Uranium	
U-235	0.2
U-238	99.8

TABLE F4.13-3

Multiplication Constant Calculation Definitions

		Number of Rods		
k-eff	BOC/EOC	Fully Withdrawn	Fully Inserted	Partial Insertion*
K_1	EOC	N		
K_2	EOC		N	
K_3	BOC	N		
K_4	BOC		N	
K_5	BOC			N
K_6	BOC	1	N-1	
K_7	EOC	1	N-1	
K_8	BOC	1		N-1
K_9	BOC		1	N-1
K_{10}	EOC	N-1	1	

*The insertion depth was assumed to be that needed to compensate the burnup reactivity swing at BOC.

FIGURE F4.13-1

SCHEMATIC OF NUCLEAR DESIGN EVALUATION PROCESS

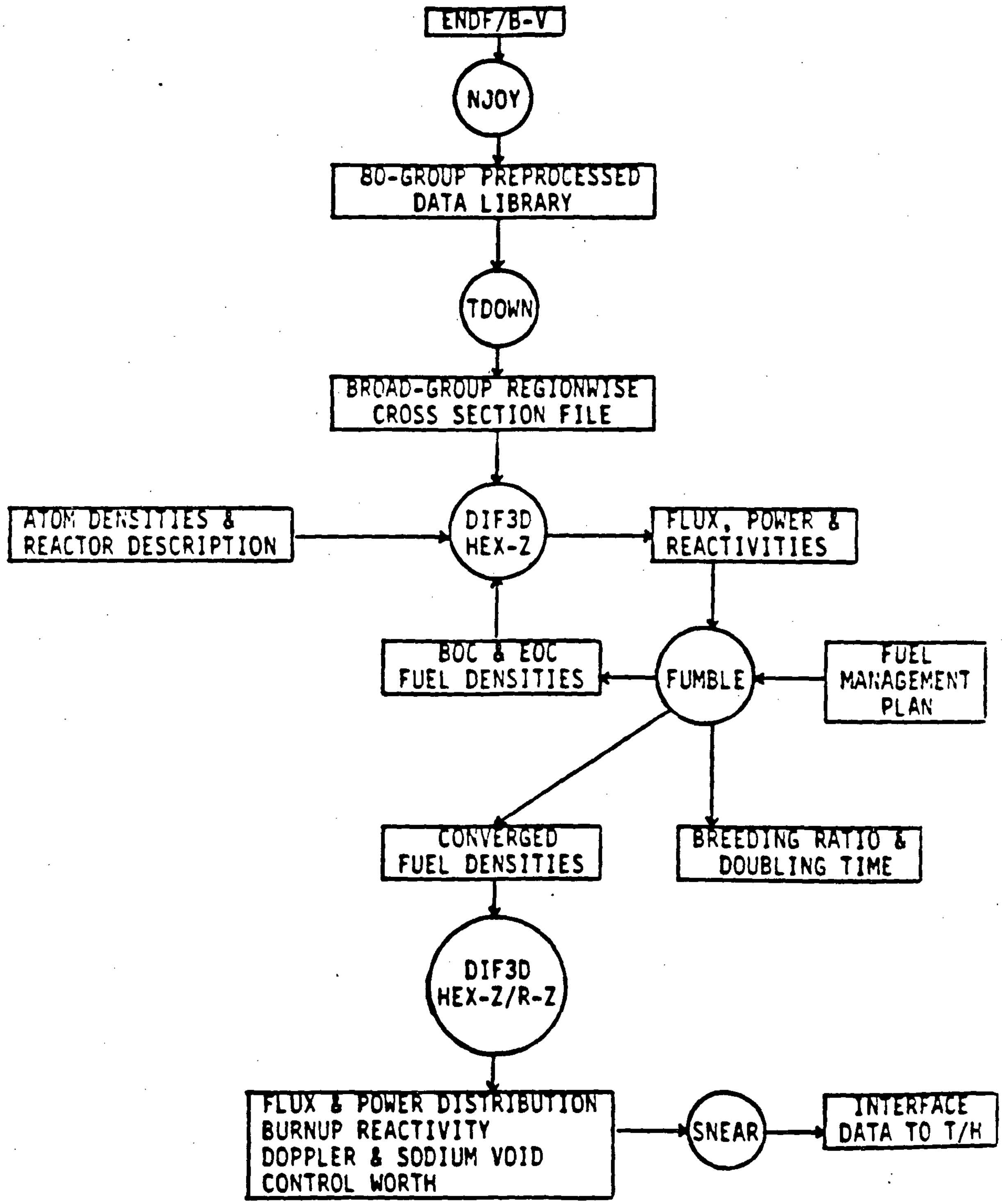
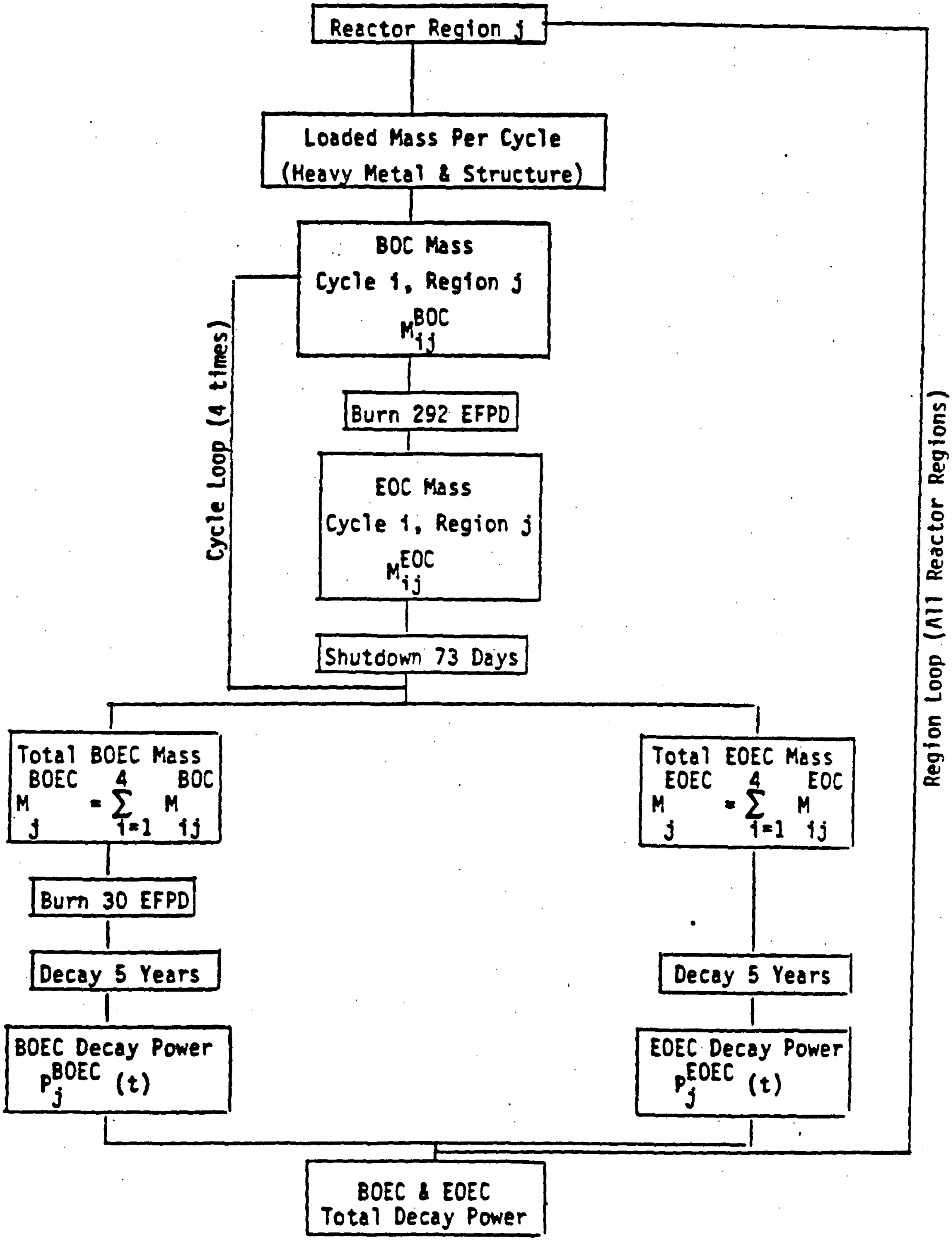


FIGURE F4.13-2

SCHEMATIC OF ORIGEN-2 DECAY HEAT CALCULATION



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F4.13-9. T. R. Falkner and R. Protsik, "Users Manual for SNPERT: A Two-Dimensional Perturbation Theory Neutronics Code," NEDE-14085, December 1975.

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4.14 Comment

Have reactivity perturbations due to seismic been considered? Can changes in core geometry hinder or prevent rod insertions for OBE or SSE?

Response

Reactivity perturbations due to seismic were evaluated for an oxide PRISM core in FY85 and found to produce less severe reactivity than the single rod withdrawal.

Due to the small core size, small number of assemblies and the small core restraint system gaps, very low impact loads are created by an SSE. The oxide core analysis by GSCRAP showed that up to about 1.0 g acceleration at the core support could be withstood without the expectation of assembly duct (load pad elevation) damage. These evaluations thus indicate that up to 1.0 g at the core support would not be expected to hamper control insertion. This acceleration is far greater than that associated with the SSE (0.3g).

Interference between scram latch release and the assembly will not occur for the SSE because the latch moving components are protected by an outer shroud tube and at all times are enclosed within the assembly as shown in Figures 4.5-1 and 4.5-2 of the PSID. Motion past the top of the assemblies is not required for drop scram. Drive-in scram can overcome considerable resistance to scram with a powerful motor that can overcome unanticipated contact friction between the driveline and assembly. Under the drop scram mode, single or two point contact between the poison assembly and the duct may occur under SSE conditions. These momentary contacts will increase the scram times slightly. Analysis of this effect was done for the CRBRP secondary rods and found to be inconsequential.

4.15 Comment

Regarding reactivity parameters cited in the PSID, describe for the metal fuel proposed:

a) How is the Doppler constant ρ_D defined?
b) Are core radial expansion and bowing reactivity combined into a single reactivity? If so, how was this done?
c) How is the reactor vessel expansion treated? Is the wall temperature calculation an integral part of the plant transient simulation?
d) Provide a discussion of reactivity uncertainties and the potential impact on ATWS event.

4.15 Response

Doppler coefficients (Tdk/dT) are given by:

$$[k(T_2) - k(T_1)]/\ln(T_2/T_1),$$

where T_2 and T_1 are the temperatures of the cross section sets used in the flux computations. Normally Doppler calculations are performed for various fuel zones to separate the Doppler coefficients for the driver fuel, internal blanket, axial blanket, and radial blanket. As a general practice, global Doppler calculations are carried out using fine group structure (for instance, 22 groups) to minimize the sensitivity of group structure.

Radial thermal expansion and bowing are combined into a single value representing the radial growth (or shrinkage) of the central plane of the core. This radial size change is applied to calculate the radial reactivity feedback.

The radial expansion coefficients (Rdk/dR), for constant material mass are computed by uniformly increasing the core size by a constant fraction (e.g., 5 or 10 percent) without changing the material mass. That is, material isotopic densities are reduced accordingly. Then,

$$Rdk/dR = \frac{k(\text{radial expanded}) - k(\text{reference})}{\text{radial expansion fraction}}$$

The expansion coefficient represents the total effect when expanding the core size, and includes two effects of opposite sign - a positive leakage feedback due to reduction of neutron leakage and a negative density feedback due to reduction of fuel densities. Thus, the leakage components of the expansion coefficients can be derived as follows (using $d\rho/\rho = -dH/H - 2\ dR/R$):

$$(Rdk/dR)_{leakage} = (Rdk/dR)_{total} + 2(\rho dk/d\rho).$$

Vessel axial expansion is treated within reactor transient analysis by the plant system transient simulation codes, ARIES and SASSYS. Vessel expansion effects control rod insertion into the core, and, along with control driveline expansion, are directly modelled in these codes.

The more detailed core transient analysis code, CORTAC-2D, uses coolant inlet temperature and driveline position data generated by a reactor transient code and does not directly estimate vessel expansion and its feedback. These are input as boundary conditions as part of the transient definition.

Reactivity uncertainties and effects are discussed in the response to Comment 4.16. Uncertainties are divided into two categories or situations. The first represents "large" uncertainties that should be applied in determining the control system worth requirement. Large margins are prudent in conceptual design to ensure adequate worth in the control system.

4.15 Response (Continued)

The second situation relates to the uncertainties in a "known, operated" core. These represent best estimates of uncertainties for which excess reactivity should be designed into the core to permit critical operation over the full cycle. Because the core (or a replicate core) has been tested, many calculational uncertainties are reduced compared to the "large" uncertainties applied in conceptual design of worth requirements. Also, reload fuel fissile specification by batch reduces the excess reactivity to the minimum required for anticipated operation and fabrication variations. Excess reactivity and burnup swing are minimized for low rod worths based on withdrawal from full power banked positions. The resulting unprotected transient overpower events are thus reduced in severity so that core equilibrium power and temperature from the transient overpower without scram U/TOP do not exceed acceptable limits. With reduced excess reactivity and burnup swing, the core is able to accommodate an "all rods out" U/TOP without fuel melting, coolant boiling or early pin failures on a nominal basis.

4.16 Comment

What are the uncertainties in the control rod worth? Are systematic errors taken into account?

Response

The uncertainties in the control rod worth are discussed in Section 4.3.2.5 of the PSID and summarized in Table 4.3-9. Since issue of the PSID, the uncertainty estimates for a known, operating core have been recalculated. The uncertainty values in Table 4.3-9 have been changed to read as follows.

	Uncertainty for Use in Estimated Control System Design Worth	Uncertainty for Use Estimated Nth Core Operational Worth
Temperature Defect	(0.29)	(0.15$)
Criticality	(1.00$)	(0.10$)
Fissile Enrichment	(1.00$)	(0.20$)
Fuel Management	(1.00$)	(0.10$)
Combined Uncertainty*	(1.76$)	(0.29$)

*Statistically independent combined uncertainty

This treatment of control rod worth uncertainties follows that of the Clinch River Project. We are aware of no systematic errors other than those inherent in the design analysis calculational tools, which will be corrected by calibration against experimental test data.

4.17 Comment

Provide material properties for HT-9; density, linear expansion coefficient, microscopic total cross-section (averaged over the reactor spectrum), and its composition. Describe expected changes in these properties with exposure. Also, provide a typical predicted flux spectrum.

Response

See the data on the following 10 pages.

TC-293 ALLOY PROPERTIES DATA BOOK	ALLOY: HT-9
REVISION: 5, 1/18/81	PROPERTY: NOMINAL COMPOSITION

Nominal Composition, Weight Percent

Fe	Cr	Ni	Mo	W	V	Si	Mn	C
85.0	12.0	0.5	1.0	0.5	0.3	0.25	0.2	0.20

Nominal Composition, Atom Percent

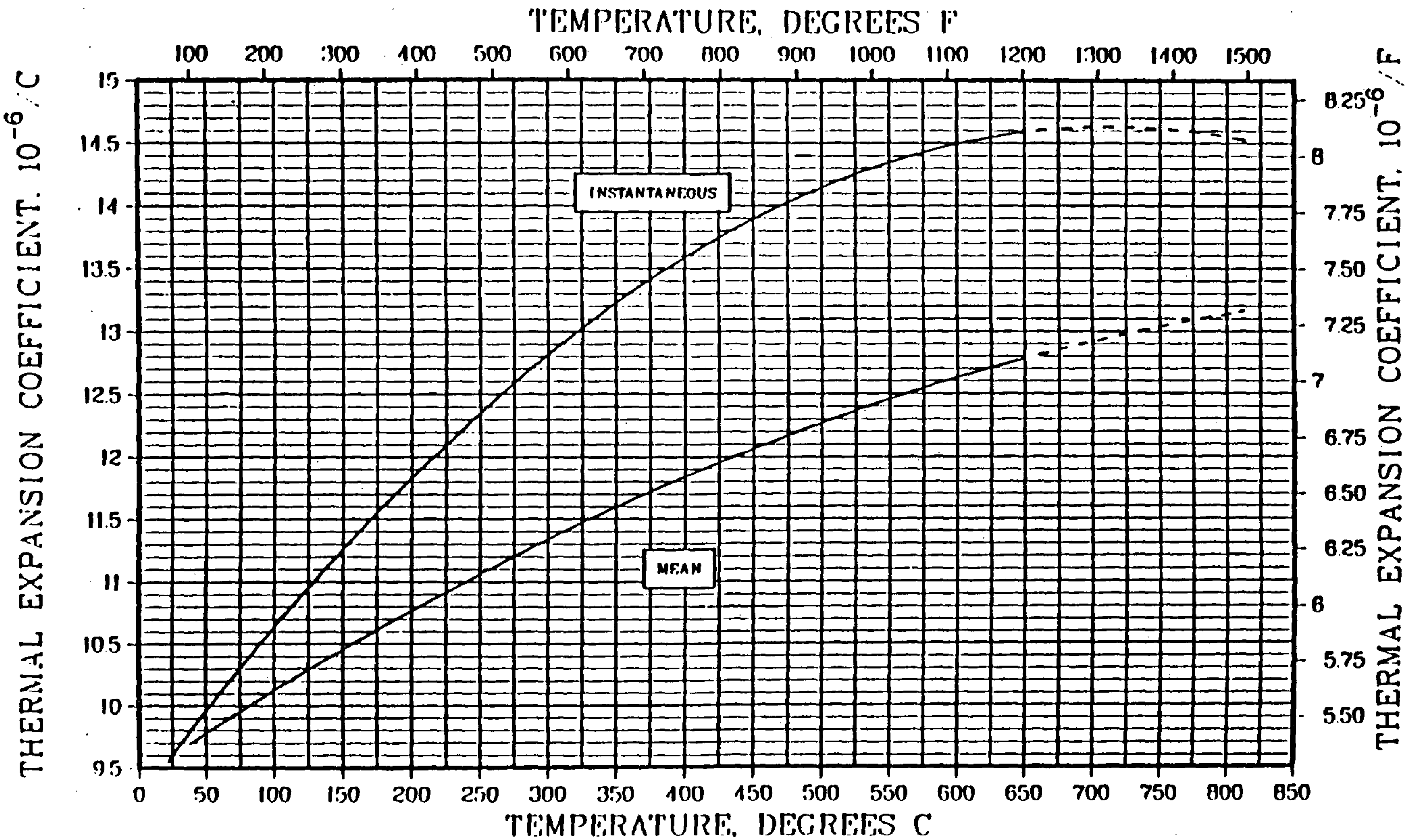
Fe	Cr	Ni	Mo	W	V	Si	Mn	C
84.1	12.7	0.5	0.6	0.2	0.3	0.5	0.2	0.9

Thermomechanical Treatment

1038°C/1 min/AC + 760°C/0.5 hr/AC

MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS

MATERIAL HT-9 STEEL	PROPERTY THERMAL EXPANSION COEFFICIENT



Thermal expansion, mean and instantaneous. See pages 1.1 and 1.2 for the equations used to generate these curves.

MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS

MATERIAL	HT-9 STEEL	PROPERTY	THERMAL EXPANSION COEFFICIENT

TABLE I – INSTANTANEOUS COEFFICIENT

BASIC UNITS SI (US UNITS IN PARENTHESES)			
TEMPERATURE		EXPANSION COEFFICIENT	
DEGREES C	DEGREES F	10^{-6}/C	10^{-6}/F
2.1110+001	(6.9998+001)	9.5360+000	(5.2978+000)
5.0000+001	(1.2200+002)	9.9522+000	(5.5290+000)
1.0000+002	(2.1200+002)	1.0631+001	(5.9059+000)
1.5000+002	(3.0200+002)	1.1256+001	(6.2532+000)
2.0000+002	(3.9200+002)	1.1828+001	(6.5711+000)
2.5000+002	(4.8200+002)	1.2347+001	(6.8594+000)
3.0000+002	(5.7200+002)	1.2813+001	(7.1182+000)
3.5000+002	(6.6200+002)	1.3228+001	(7.3476+000)
4.0000+002	(7.5200+002)	1.3585+001	(7.5474+000)
4.5000+002	(8.4200+002)	1.3892+001	(7.7177+000)
5.0000+002	(9.3200+002)	1.4145+001	(7.8584+000)
5.5000+002	(1.0220+003)	1.4345+001	(7.9697+000)
6.0000+002	(1.1120+003)	1.4493+001	(8.0515+000)
6.5000+002	(1.2020+003)	1.4587+001	(8.1037+000)
7.0000+002	(1.2920+003)	1.4628+001	(8.1265+000)
7.5000+002	(1.3820+003)	1.4615+001	(8.1197+000)
8.0000+002	(1.4720+003)	1.4550+001	(8.0834+000)
8.1560+002	(1.5001+003)	1.4518+001	(8.0661+000)

BASIC UNITS US (SI UNITS IN PARENTHESES)			
TEMPERATURE		EXPANSION COEFFICIENT	
DEGREES F	DEGREES C	10^{-6}/F	10^{-6}/C
6.9998+001	(2.1110+001)	5.2978+000	(9.5360+000)
1.0000+002	(3.7778+001)	5.4324+000	(9.7783+000)
2.0000+002	(9.3333+001)	5.8573+000	(1.0543+001)
3.0000+002	(1.4889+002)	6.2458+000	(1.1242+001)
4.0000+002	(2.0444+002)	6.5979+000	(1.1876+001)
5.0000+002	(2.6000+002)	6.9135+000	(1.2444+001)
6.0000+002	(3.1556+002)	7.1927+000	(1.2947+001)
7.0000+002	(3.7111+002)	7.4355+000	(1.3384+001)
8.0000+002	(4.2667+002)	7.6418+000	(1.3755+001)
9.0000+002	(4.8222+002)	7.8118+000	(1.4061+001)
1.0000+003	(5.3778+002)	7.9452+000	(1.4301+001)
1.1000+003	(5.9333+002)	8.0423+000	(1.4476+001)
1.2000+003	(6.4889+002)	8.1029+000	(1.4585+001)
1.3000+003	(7.0444+002)	8.1271+000	(1.4629+001)
1.4000+003	(7.6000+002)	8.1148+000	(1.4607+001)
1.5001+003	(8.1560+002)	8.0661+000	(1.4519+001)

The curve on page 10 and the values above were calculated from the following equation:

$$\alpha = 9.2207 + 1.5161\times10^{-2}T - 1.0624\times10^{-5}T^2$$, where

α = Instantaneous Coefficient, 10^{-6}/C
T = Temperature, Degrees C $(21 < T < 650)$

The uncertainty is estimated to be ±10%. Multiply US units by 1.8 to convert to SI units.

MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS

MATERIAL HT-9 STEEL	PROPERTY THERMAL EXPANSION COEFFICIENT

TABLE II – MEAN COEFFICIENT

BASIC UNITS SI (US UNITS IN PARENTHESES)				BASIC UNITS US (SI UNITS IN PARENTHESES)			
TEMPERATURE		EXPANSION COEFFICIENT		TEMPERATURE		EXPANSION COEFFICIENT	
DEGREES C	DEGREES F	10^{-6}/C	10^{-6}/F	DEGREES F	DEGREES C	10^{-6}/F	10^{-6}/C
3.7770+001	(9.9986+001)	9.6859+000	(5.3811+000)	9.9986+001	(3.7770+001)	5.3811+000	(9.6859+000)
5.0000+001	(1.2200+002)	9.7738+000	(5.4299+000)	1.0000+002	(3.7778+001)	5.3811+000	(9.6860+000)
1.0000+002	(2.1200+002)	1.0122+001	(5.6232+000)	2.0000+002	(9.3333+001)	5.5980+000	(1.0076+001)
1.5000+002	(3.0200+002)	1.0452+001	(5.8067+000)	3.0000+002	(1.4889+002)	5.8028+000	(1.0445+001)
2.0000+002	(3.9200+002)	1.0765+001	(5.9804+000)	4.0000+002	(2.0444+002)	5.9954+000	(1.0792+001)
2.5000+002	(4.8200+002)	1.1060+001	(6.1443+000)	5.0000+002	(2.6000+002)	6.1759+000	(1.1117+001)
3.0000+002	(5.7200+002)	1.1337+001	(6.2983+000)	6.0000+002	(3.1556+002)	6.3442+000	(1.1420+001)
3.5000+002	(6.6200+002)	1.1596+001	(6.4425+000)	7.0000+002	(3.7111+002)	6.5004+000	(1.1701+001)
4.0000+002	(7.5200+002)	1.1838+001	(6.5768+000)	8.0000+002	(4.2667+002)	6.6444+000	(1.1960+001)
4.5000+002	(8.4200+002)	1.2062+001	(6.7013+000)	9.0000+002	(4.8222+002)	6.7263+000	(1.2107+001)
5.0000+002	(9.3200+002)	1.2269+001	(6.8160+000)	1.0000+003	(5.3778+002)	6.8961+000	(1.2413+001)
5.5000+002	(1.0220+003)	1.2457+001	(6.9208+000)	1.1000+003	(5.9333+002)	7.0037+000	(1.2607+001)
6.0000+002	(1.1120+003)	1.2628+001	(7.0158+000)	1.2000+003	(6.4889+002)	7.0992+000	(1.2778+001)
6.5000+002	(1.2020+003)	1.2782+001	(7.1009+000)	1.3000+003	(7.0444+002)	7.1825+000	(1.2928+001)
7.0000+002	(1.2920+003)	1.2917+001	(7.1763+000)	1.4000+003	(7.6000+002)	7.2537+000	(1.3057+001)
7.5000+002	(1.3820+003)	1.3035+001	(7.2417+000)	1.5001+003	(8.1560+002)	7.3127+000	(1.3163+001)
8.0000+002	(1.4720+003)	1.3135+001	(7.2974+000)				
8.1560+002	(1.5001+003)	1.3163+001	(7.3127+000)				

The curve on page 10 and the values above were calculated from the following equation:

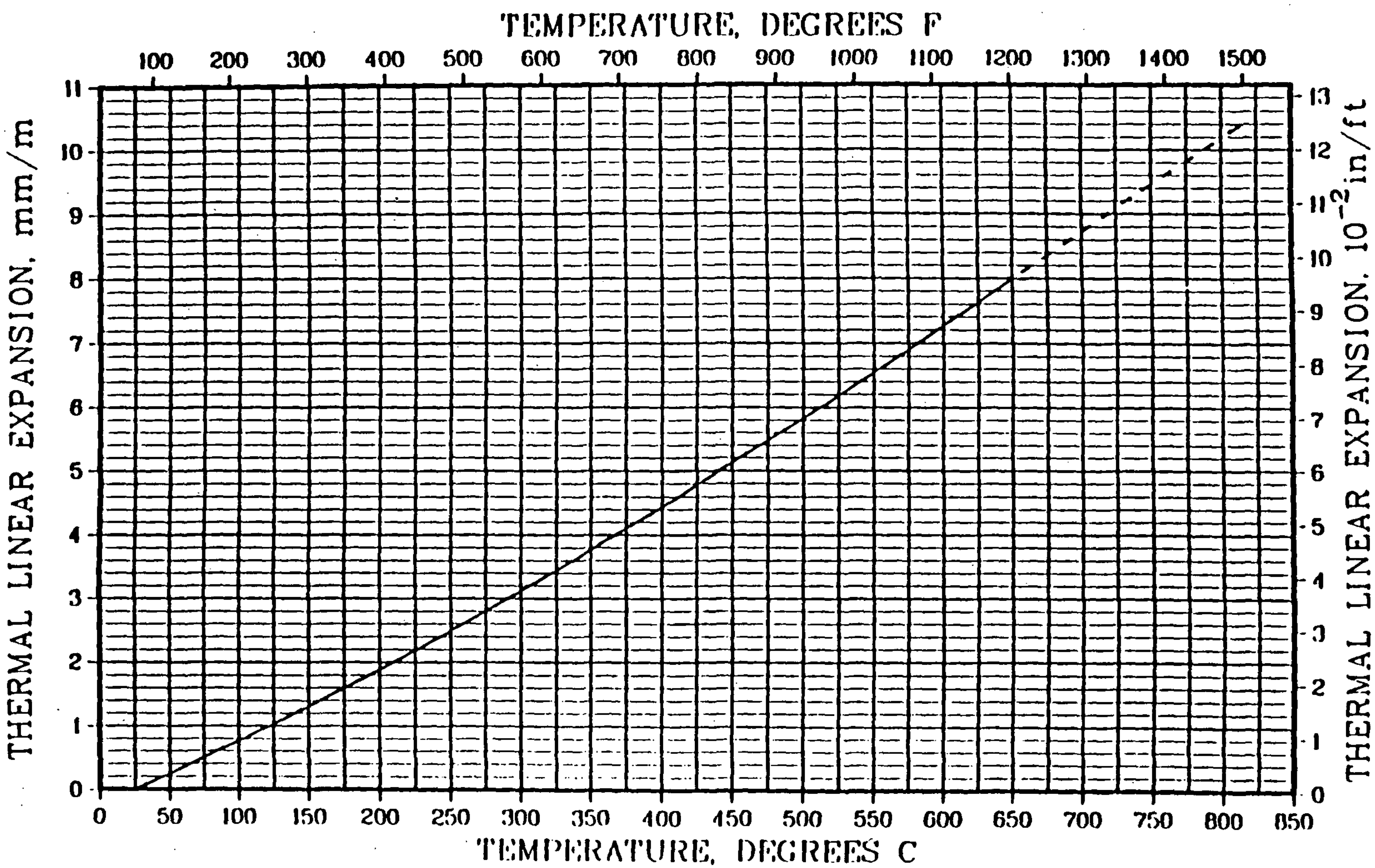
$$\alpha = -235.20/R + 9.2207T/R + 7.5806\times10^{-3}T^2/R - 3.5412\times10^{-6}T^3/R, \text{ where}$$

α = Mean Coefficient, 10^{-6}/C
R = T-25
T Temperature, Degrees C (37 < T < 650)

The uncertainty is estimated to be ±5%. Multiply US units by 1.8 to convert to SI units.

MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS

MATERIAL HT-9 STEEL	PROPERTY LINEAR THERMAL EXPANSION



1 thermal expansion. See page 2.1 for the equation used to generate th ve.

MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS

MATERIAL HT-9 STEEL	PROPERTY LINEAR THERMAL EXPANSION

BASIC UNITS SI (US UNITS IN PARENTHESES)			
TEMPERATURE		THERMAL EXPANSION	
DEGREES C	DEGREES F	mm/m	10^{-2}in/ft
2.1110+001	(6.9998+001)	.0000	(.0000)
5.0000+001	(1.2200+002)	2.4434-001	(2.9321-001)
1.0000+002	(2.1200+002)	7.5913-001	(9.1096-001)
1.5000+002	(3.0200+002)	1.3065+000	(1.5678+000)
2.0000+002	(3.9200+002)	1.8838+000	(2.2606+000)
2.5000+002	(4.8200+002)	2.4884+000	(2.9861+000)
3.0000+002	(5.7200+002)	3.1177+000	(3.7412+000)
3.5000+002	(6.6200+002)	3.7688+000	(4.5226+000)
4.0000+002	(7.5200+002)	4.4393+000	(5.3272+000)
4.5000+002	(8.4200+002)	5.1265+000	(6.1518+000)
5.0000+002	(9.3200+002)	5.8278+000	(6.9032+000)
5.5000+002	(1.0220+003)	6.5401+000	(7.8482+000)
6.0000+002	(1.1120+003)	7.2613+000	(8.7136+000)
6.5000+002	(1.2020+003)	7.9886+000	(9.5863+000)
7.0000+002	(1.2920+003)	8.7192+000	(1.0463+001)
7.5000+002	(1.3820+003)	9.4505+000	(1.1341+001)
8.0000+002	(1.4720+003)	1.0180+001	(1.2216+001)
8.1560+002	(1.5001+003)	1.0407+001	(1.2488+001)

BASIC UNITS US (SI UNITS IN PARENTHESES)			
TEMPERATURE		THERMAL EXPANSION	
DEGREES F	DEGREES C	10^{-2}in/ft	mm/m
6.9998+001	(2.1110+001)	.0000	(.0000)
1.0000+002	(3.7778+001)	1.4852-001	(1.2377-001)
2.0000+002	(9.3333+001)	8.2627-001	(6.8855-001)
3.0000+002	(1.4889+002)	1.5528+000	(1.2940+000)
4.0000+002	(2.0444+002)	2.3238+000	(1.9365+000)
5.0000+002	(2.6000+002)	3.1349+000	(2.6124+000)
6.0000+002	(3.1556+002)	3.9816+000	(3.3180+000)
7.0000+002	(3.7111+002)	4.8597+000	(4.0497+000)
8.0000+002	(4.2667+002)	5.7647+000	(4.8039+000)
9.0000+002	(4.8222+002)	6.6923+000	(5.5769+000)
1.0000+003	(5.3778+002)	7.6381+000	(6.3651+000)
1.1000+003	(5.9333+002)	8.5977+000	(7.1648+000)
1.2000+003	(6.4889+002)	9.5668+000	(7.9723+000)
1.3000+003	(7.0444+002)	1.0541+001	(8.7842+000)
1.4000+003	(7.6000+002)	1.1516+001	(9.5966+000)
1.5001+003	(8.1560+002)	1.2488+001	(1.0407+001)

The curve on page 20 and the values above were calculated from the following equation:

$$\Delta L/L = -2.3520\cdot10^{-1} + 8.2207\cdot10^{-3}T + 7.5806\cdot10^{-6}T^{2} - 3.5412\cdot10^{-9}T^{3}, \text{ where:}$$

$\Delta L/L$ = Strain, mm/m

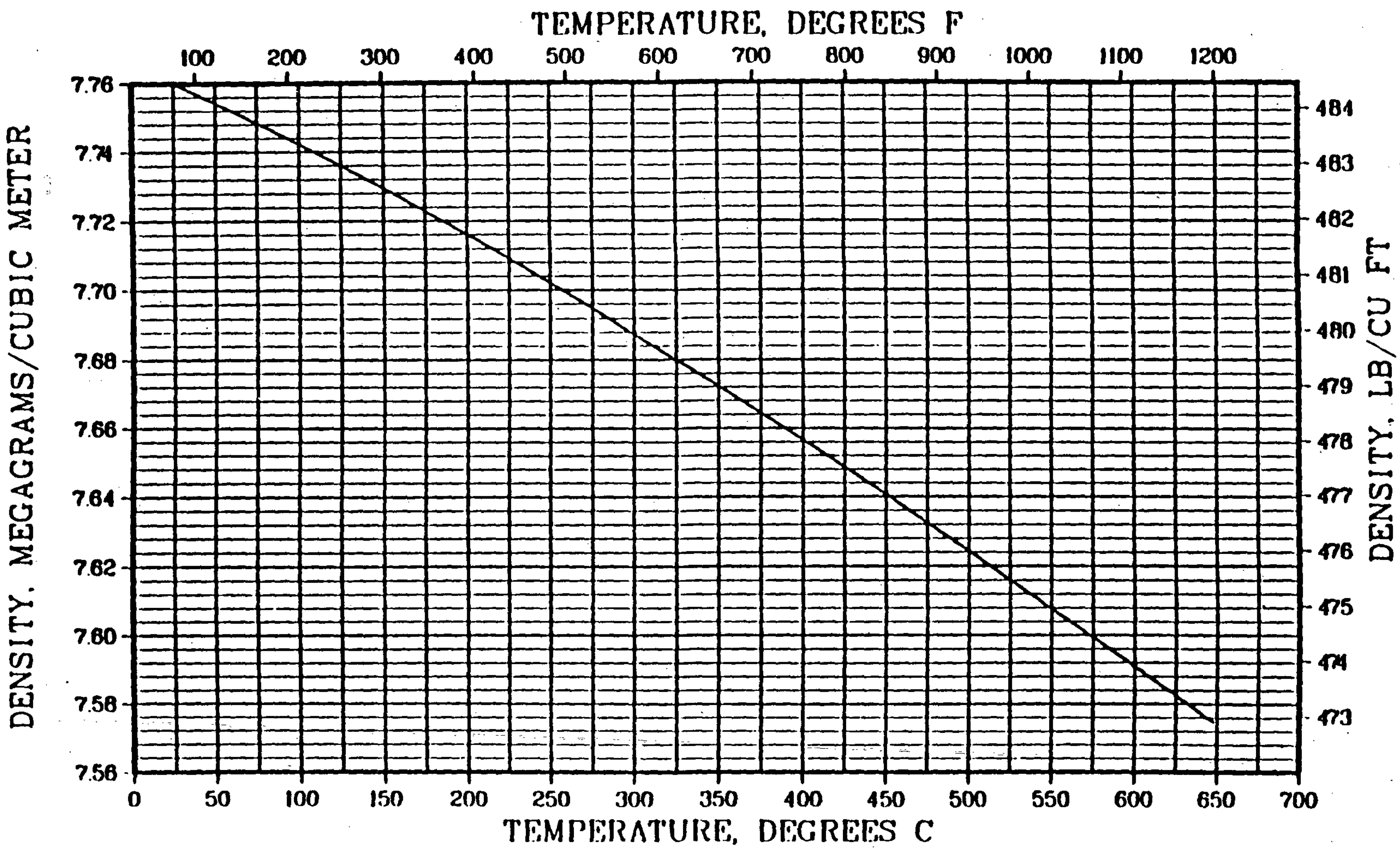
T = Temperature, Degrees C (21 < T < 650)

The uncertainty is estimated to be ±5%. Multiply US units by 0.8333334 to convert to SI units.

RESPONSES TO NRC COMMENTS ON PRISM PSID CHAPTER 4

MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS

MATERIAL HT-9 STEEL	PROPERTY DENSITY (GRAVIMETRIC)



.sity (Gravimetric). See page 1.1 for the equation used to generate this

MATERIALS HANDBOOK FOR FUSION ENERGY SYSTEMS

MATERIAL HT-9 STEEL	PROPERTY DENSITY (GRAVIMETRIC)

BASIC UNITS SI (US UNITS IN PARENTHESES)			
TEMPERATURE		DENSITY	
DEGREES C	DEGREES F	Mg/m³	lb/ft³
2.5000+001	(7.7000+001)	7.7600+000	(4.8445+002)
5.0000+001	(1.2200+002)	7.7543+000	(4.8410+002)
7.5000+001	(1.6700+002)	7.7484+000	(4.8373+002)
1.0000+002	(2.1200+002)	7.7423+000	(4.8335+002)
1.2500+002	(2.5700+002)	7.7360+000	(4.8296+002)
1.5000+002	(3.0200+002)	7.7296+000	(4.8256+002)
1.7500+002	(3.4700+002)	7.7229+000	(4.8214+002)
2.0000+002	(3.9200+002)	7.7161+000	(4.8172+002)
2.2500+002	(4.3700+002)	7.7092+000	(4.8128+002)
2.5000+002	(4.8200+002)	7.7021+000	(4.8084+002)
2.7500+002	(5.2700+002)	7.6948+000	(4.8039+002)
3.0000+002	(5.7200+002)	7.6874+000	(4.7992+002)
3.2500+002	(6.1700+002)	7.6799+000	(4.7945+002)
3.5000+002	(6.6200+002)	7.6723+000	(4.7898+002)
3.7500+002	(7.0700+002)	7.6645+000	(4.7849+002)
4.0000+002	(7.5200+002)	7.6567+000	(4.7800+002)
4.2500+002	(7.9700+002)	7.6487+000	(4.7751+002)
4.5000+002	(8.4200+002)	7.6407+000	(4.7700+002)
4.7500+002	(8.8700+002)	7.6325+000	(4.7650+002)
5.0000+002	(9.3200+002)	7.6243+000	(4.7599+002)
5.2500+002	(9.7700+002)	7.6161+000	(4.7547+002)
5.5000+002	(1.0220+003)	7.6077+000	(4.7495+002)
5.7500+002	(1.0670+003)	7.5994+000	(4.7443+002)
6.0000+002	(1.1120+003)	7.5910+000	(4.7390+002)
6.2500+002	(1.1570+003)	7.5825+000	(4.7337+002)
6.5000+002	(1.2020+003)	7.5740+000	(4.7284+002)

BASIC UNITS US (SI UNITS IN PARENTHESES)			
TEMPERATURE		DENSITY	
DEGREES F	DEGREES C	lb/ft³	Mg/m³
7.7000+001	(2.5000+001)	4.8445+002	(7.7600+000)
1.0000+002	(3.7778+001)	4.8428+002	(7.7571+000)
1.5000+002	(6.5556+001)	4.8387+002	(7.7507+000)
2.0000+002	(9.3333+001)	4.8345+002	(7.7440+000)
2.5000+002	(1.2111+002)	4.8302+002	(7.7370+000)
3.0000+002	(1.4889+002)	4.8257+002	(7.7299+000)
3.5000+002	(1.7667+002)	4.8211+002	(7.7225+000)
4.0000+002	(2.0444+002)	4.8164+002	(7.7149+000)
4.5000+002	(2.3222+002)	4.8116+002	(7.7071+000)
5.0000+002	(2.6000+002)	4.8066+002	(7.6992+000)
5.5000+002	(2.8778+002)	4.8015+002	(7.6911+000)
6.0000+002	(3.1556+002)	4.7963+002	(7.6828+000)
6.5000+002	(3.4333+002)	4.7911+002	(7.6743+000)
7.0000+002	(3.7111+002)	4.7857+002	(7.6657+000)
7.5000+002	(3.9889+002)	4.7802+002	(7.6570+000)
8.0000+002	(4.2667+002)	4.7747+002	(7.6482+000)
8.5000+002	(4.5444+002)	4.7691+002	(7.6392+000)
9.0000+002	(4.8222+002)	4.7635+002	(7.6302+000)
9.5000+002	(5.1000+002)	4.7578+002	(7.6210+000)
1.0000+003	(5.3778+002)	4.7520+002	(7.6118+000)
1.0500+003	(5.6556+002)	4.7462+002	(7.6025+000)
1.1000+003	(5.9333+002)	4.7404+002	(7.5932+000)
1.1500+003	(6.2111+002)	4.7346+002	(7.5838+000)
1.2020+003	(6.5000+002)	4.7284+002	(7.5740+000)

The curve on page 10 and the values above were calculated from the following equation:

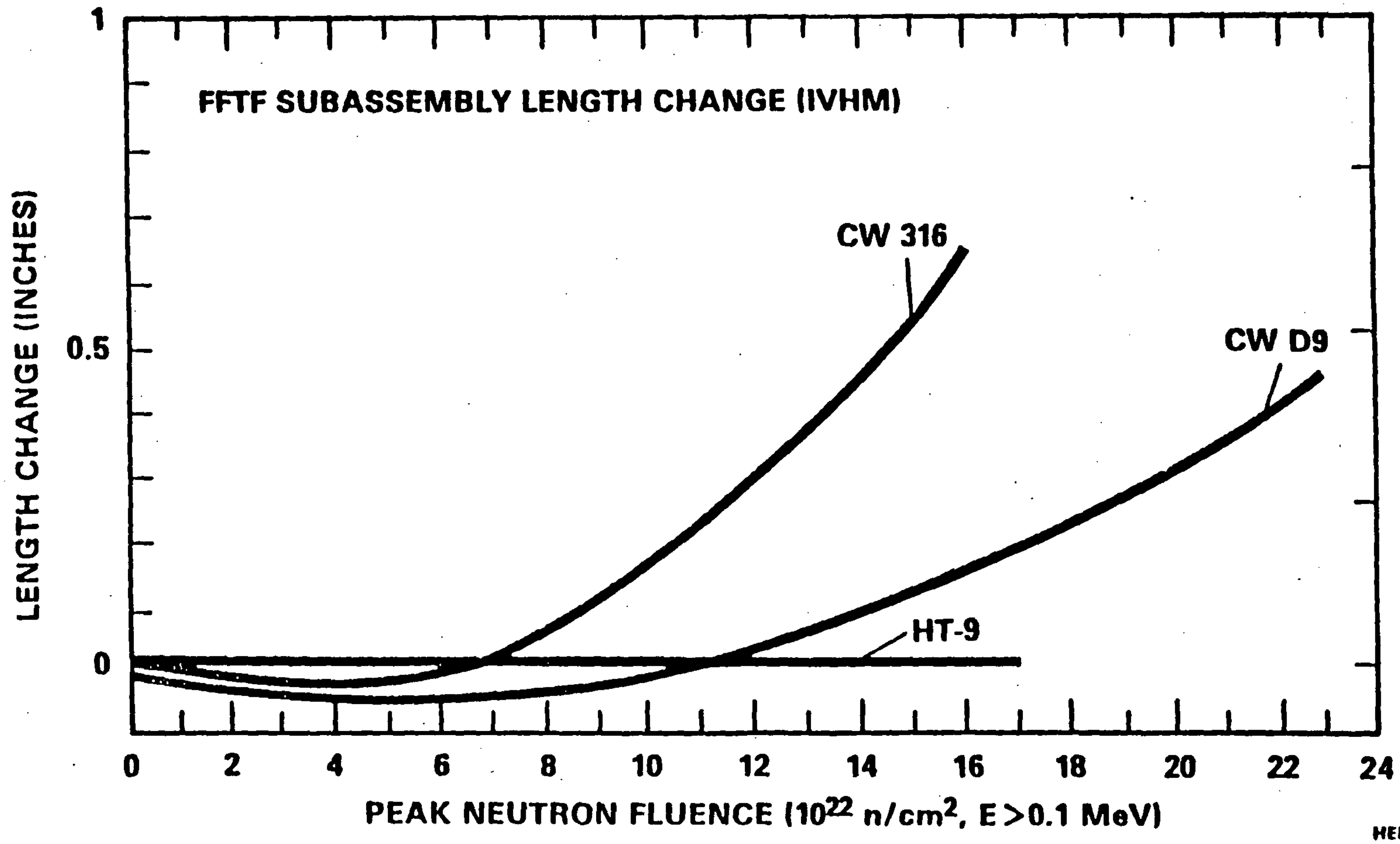
$\rho = 7.760[1-3(-2.3520\times10^{-4}+9.2207\times10^{-6}T+7.5808\times10^{-9}T^2-3.5412\times10^{-12}T^3)]$, where

ρ=Gravimetric Density, Megagrams/Cubic Meter
T=Temperature, Degrees C $(25 < T < 650)$

Multiply US units by 0.016018 to convert to SI units.

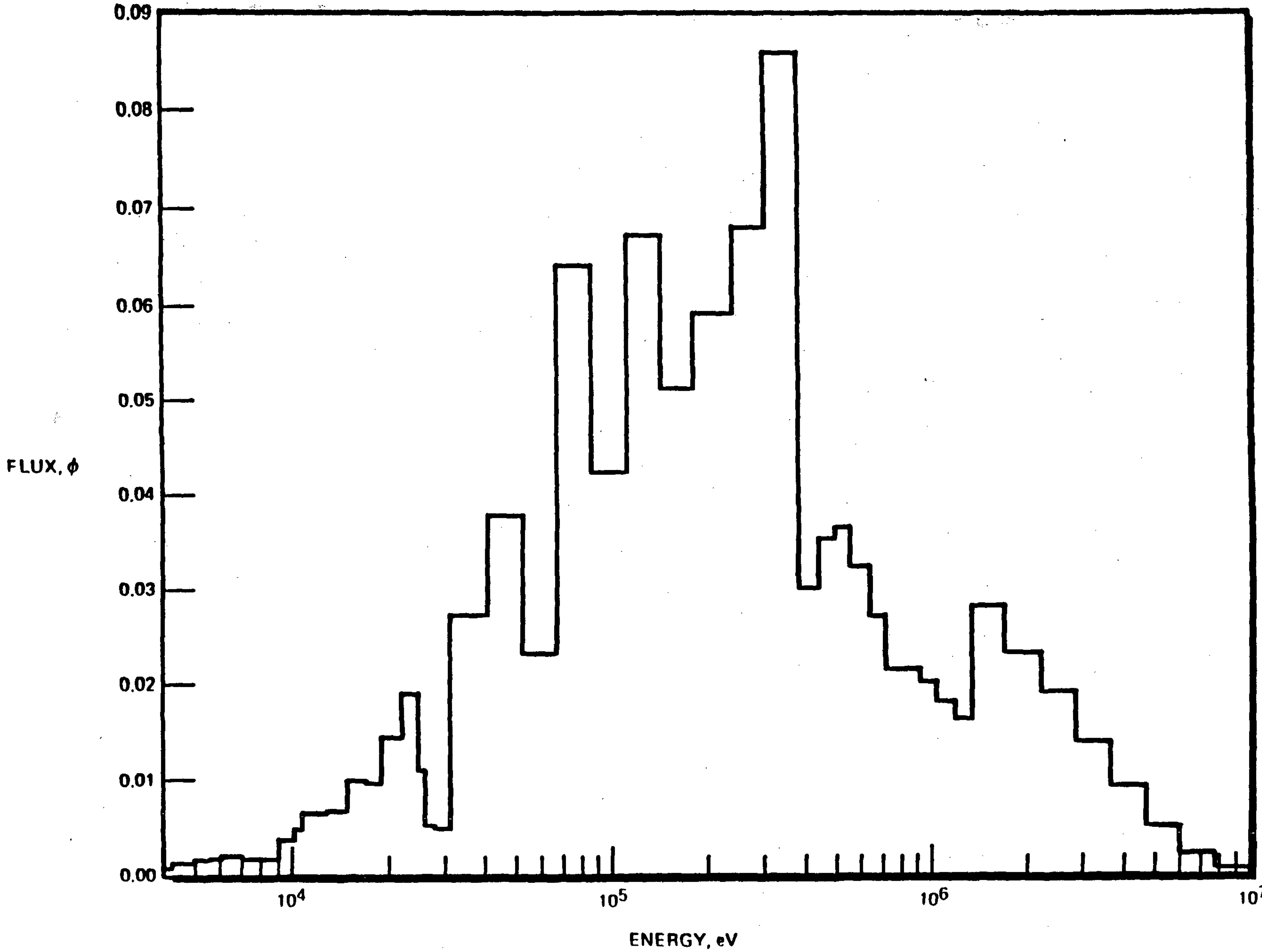
HEDL

FULL-SIZE COMPONENT EXPERIENCE CONFIRMS ALLOY SELECTION



HEDL 8602-140

TYPICAL PREDICTED FLUX SPECTRUM



87-232-01

4.18 Comment

Provide the philosophy and a description of the validation of your design methods (computer codes) and data base. (A series of references would suffice).

Response

The attached table provides a list of the computer codes being used for core design analysis and their validation status. Most of the codes are LMR national standard codes, either developed and validated (or being validated) by a national laboratory (e.g., SASSYS by ANL) or by a national committee under DOE direction (e.g., the LIFE series of fuel pin analysis codes). A few are GE design tools that were developed, and in many cases validated, for use on the Clinch River Project.

The attempt has been made to utilize, to the maximum extent possible, national standard codes for design methods, particularly those that will be used for safety analyses. All the codes listed can be considered verified, that is, their numerics have been checked and found to be correct. Many of the codes require specific validation to the PRISM reactor. This validation activity is planned as part of the supporting R&D program and as part of the PRISM Safety Test.

Core Design Computer Codes and Validation Status

Code	Validation Status
Nuclear Analysis	
SN2D, FUMBLE, SNEAR, DIF3D, SN-PERT, ORIGEN-II	(National codes developed by national laboratories; see References in answer to Question I-13)
Fuel/Blanket Pin Analysis:	
LIFE-M	LIFE code series is national standard; oxide version fully validated, metal fuel version being validated by ANL
Absorber Pin Analysis	
CONROD	GE code, Spec 23A3087 (1984); additional validation in progress by HEDL
Assembly and Core T/H Analysis	
NELI	GE internal code, validation incomplete
ORIFORT	GE code, ARSD-00104 (1982)
FLODISC	HEDL code, HEDL-TC-874 (1977)
CORTEN	GE code, Spec 23A3063 (1984)
FULMIX	GE code, Spec 23A3051 (1983)
FUELTEMP	GE internal code; not validated
COBRA-WC	Developed by PNL; for validation see PNL-4141 (1982) and GEFR-00726 (1984)
COMMIX-1A	Developed by ANL; for validation see ANL-82-25 (1983) GEFR-00724 (1984) GEFR-00720 (1984)
Core System Transient Analysis	
CORTAC	GEAP-14115 (1976); not fully validated
Reactor System Transient Analysis	
SASSYS	National code; currently being validated by ANL
ARIES	GE design code; validated by comparison to SASSYS and EBR-II test data
Assembly Elastic/Inelastic Stress Analysis	
NUBOW-3D	National code; validation by ANL
ANSYS	Swanson Analysis System, Inc.
BUNDUCT	GE assembly lifetime analysis code; not validated
DUKSHAP	GE linking code between NUBOW and CORTAC; not validated
Core Seismic Analysis	
GSCRAP	GE version of national standard SCRAP; not validated
Refueling Loads/Interference Analysis	
DIAS	GE code to check geometry interferences; not validated

4.19 Comment

Please explain the Core Special Assemblies (CSA) referred to in Chapter 4 of the PSID.

Response

Core Special Assemblies (CSAs) are special purpose assemblies that are not part of normal power operation. During pre-operational tests, Core Filter Assemblies (CFAs) are substituted for the core assemblies in order to filter the primary coolant. These assemblies are then removed from the reactor to remove the trapped debris. CSA's may also contain instrumentation used for the pre-operational tests.

During the startup of the initial core, the neutron flux during subcritical operations may be too small to permit timely monitoring of fuel loading operations or during the first rise to critical. In these events, Neutron Source Assemblies (NSAs) are substituted for one or more core assemblies. Following an initial, short power run to build up fission products in the fuel to act as neutron sources, the NSAs are removed.

4.20 Comment

On Page 4.3-19, Table 4.3-11, justify the shutdown margin of \$1.00 rather than 1% $\Delta k/k$. The shutdown margin-requirement should be sufficient to maintain the core subcritical in the event of an inadvertent withdrawal of the highest worth control rod. The precision of the reativity swing and control rod worths seems to be overstated, i.e., not enough uncertainty is allowed, using a \$1.00 margin. What is the criterion for the single rod shutdown since it is the same as the full rod shutdown (\$0.93 in Table 4.3-11)?

Response

The reactivity criticality prediction uncertainty is an assumed value for conceptual design. Later more detailed analyses may cause a change to this value. An LMR will be expected to have a more accurate prediction of k_{eff} than an LWR because of the close-coupled nature of a fast spectrum core.

The criteria for the single rod shutdown is a nominal cold shutdown; that is, the insertion of a single rod must cover the temperature defect and the burnup swing and bring the core to a cold shutdown condition, without covering any uncertainties or providing any margin. Table 4.3-11 shows the single rod shutdown to have a margin of \$0.93.

The criteria for an n-1 rod shutdown includes, in addition, coverage of uncertainties and excess reactivity (inadvertent withdrawal of highest worth control rod) and a \$1.00 margin.

4.21 Comment

The fuel pin cladding is noted to be HT9. What material is used for the wire wrap and how is it attached? Considering the relatively large rod vibration forces, describe the effect on wire deformation and detachment along with the effect of wire deformation and detachment on both normal operating and accident conditions.

G. F. Schultheiss, GKSS-F of Germany, describes blockage formation in Nuclear Engineering and Design, 100(1987)427-433. Using results from ABACUS, GKSS, and EBR-II tests, he concludes that wire wrapping has some advantages for coolability compared to spacer grids when debris build-up occurs. However, debris detachment, high quality standards for fuel element production and failure detection are important. Describe the precautions to prevent or detect debris and its build-up. Describe the effects on fuel temperature canning in the event of debris build-up and blockage under normal operating and accident conditions. Describe the likelihood of debris blockage formation.

Response

The wire wrap material is HT9. HT9 is used for all assembly and pin structural components. While design details have not been determined, it is assumed the wire is spiraled around the pin cladding under tension and welded to the end plug at the bottom end and to the cladding near the top edge. Detachment of a small fraction of wires from the pins within a bundle is not a damaging situation. The spacer wires on adjacent pins continue to maintain correct pin spacing. This phenomenon was demonstrated in the development of the Partial Wireless Spacer System in which a large percentage of pins (~25%) were assembled into the fuel bundle without spacer wires.

Analyses of the fuel cladding temperature effects of small local coolant blockages within the fuel bundles have not been performed within PRISM conceptual design. In general, several layers of precautions are taken in design, construction and operation of PRISM to limit the risk of local blockages. These are discussed below:

A. Design Features

1. The inlet plenum is designed to provide numerous large and small flow paths to the inlets of each assembly's orificing module. These passages act to filter large objects from the coolant without restriction. Large passages are around the lower ends of the modules, and small passages are between the upper module sections.

2. Numerous openings into the orificing modules are provided. Some openings are in protected narrow regions between two adjacent modules, while others open into the larger triangular flow areas formed by three modules. This difference acts with the small flow path discussed in (4.26) to accommodate large object blockages without restrictions

4.21 Response (Continued)

3. Within each orificing module, the orificing plates use numerous intermediate size (<1 inch) holes to provide the appropriate flow restriction. This ensures that considerable quantities of ~1 inch particles must enter each module to constrict the core flow.

4. Each module includes axial flow upward through a narrow annulus (~.2 inch) which further filters particles prior to the assembly.

5. The fuel and blanket assemblies use 40 inch long lower end plugs which are wire wrapped along with the cladding. This long region of very low heat generation serves as a final filter for small particulates that enter the bundle but are too large to flow unimpeded through and out. (See G.F. Schultheiss, GKSS-F.R. Germany, Nuclear Engineering and Design, 100 (1987) 427-433).

B. Construction Features

1. Factory fabrication and improved (factory) quality control limit the potential for debris to be left in a reactor module. Factory cleanliness as opposed to the job site environment further reduces introduction of debris into the reactor.

2. Reactor handling horizontally and vertically and shipping (vibrations) improve the probability that debris will be discovered and removed prior to use.

C. Operational Features

1. Core Filter Assemblies are installed into the reactor prior to sodium fill. After the coolant is added, the primary coolant loop is operated for an extended period and varying flow rates. This washes the remaining loose particulates into the filter assemblies where they are trapped. The filter assemblies are removed and cleaned at intervals during this test and cleanup process. Fuel is not loaded into the reactor until cleanliness has been assured.

Additional discussions of the defenses and effects of local blockages in LMRs is contained in Chapter 15 (especially 15.4.1.3 through 15.4.1.4) of the Clinch River PSAR.

4.22 Comment

Figure 4.4-1, 30 degree sector shows coordinate 7,2 as a radial blanket assembly. The full core map shows this to be shield assembly. Which is the correct assembly? What is the effect on the core outlet temperatures and mid-wall temperatures of using a blanket instead of a shield assembly? Which assembly was used in the accident analyses and

4.22 Comment (Continued)

what is the effect? How are the changes in these two core patterns, e.g., orificing, expected to affect the Chapter 15 analyses? It is suggested that the limiting case, or cases, be reanalyzed to determine if additional must be reanalyzed.

Response

Figure 4.2-1 of the PSID is an updated view of the core layout and is correct. The thermo-hydraulics for the updated core had not been completed by the issue date of the PSID so the data from the prior generation of the core design were included as representative. Figure 4.4-1 correctly identifies this core for which analytical results were reported. The peak cladding and fuel temperatures were not strongly dependent upon the substitution of shield for blanket assemblies at the core perimeter. The BDBA performance presented in Appendix E of the PSID models the core of Figure 4.4-1; however, the substitution will not greatly affect the results.

The substitution of a radial blanket assembly for a shield assembly and the subsequent minor reorificing of the core is expected to have little effect on the Chapter 15 transient analyses.

4.23. Comment

Discuss the basis for the estimated uncertainties associated with the control rod reactivity worths and other inherent reactivity feedbacks relied upon in the safety analysis, considering factors such as the analytical tools used in their calculation, validation of those analytical tools and any experimental verification (completed or planned).

Response

The inclusion of uncertainties in conceptual design is limited to steady state and design basis evaluations. Fuel, fission gas plena, coolant and structure temperatures include the 2-sigma hot channel factors/uncertainties. See attached Table of Uncertainties and the resposne to Comment 4.2-9 Evaluations of beyond design basis accident (BDBA) events and phenomena use nominal (best engineering estimate) methods. During the current conceptual design, large margins to core failure limits are being maintained to provide some assurance that inclusion of uncertainties and more detailed analysis methods will not alter the conclusions of the evaluation.

TABLE [illegible]-1. NUCLEAR, THERMOHYDRAULIC AND PROPERTIES UNCERTAINTIES AND HOT CHANNEL FACTORS FOR FUEL ASSEMBLY CONCEPTUAL DESIGN

FUEL ASSEMBLY INTERNAL TEMPERATURES

	COOLANT TEMP RISE FOR CLAD & FUEL TEMP		FILM TEMP RISE FOR CLAD TEMP		FILM TEMP RISE FOR FUEL TEMP	
	BOL	EOL	BOL	EOL	BOL	EOL
DIRECT UNCERTAINTIES						
POWER LEVEL AND CONTROL DEADBAND	1.030	1.030	1.030	1.030	1.030	1.030
INLET FLOW DISTRIBUTION	1.020	1.020	1.000	1.000	1.000	1.000
FLOW CALCULATION	1.030	1.030	1.006	1.006	1.006	1.006
CLADDING CIRC VARIATION	1.000	1.000	1.900	1.900	1.000	1.000
PELLET ECCENTRICITY	1.000	1.000	1.140	1.000	1.000	1.000
PHYSICS MODEL AND ROD BANK	1.040	1.040	1.040	1.040	1.040	1.040
STATISTICAL UNCERTAINTIES						
INLET FLOW DISTRIBUTION	1.059	1.059	1.000	1.000	1.000	1.000
WIRE WRAP ORIENTATION	1.010	1.010	1.000	1.000	1.000	1.000
SUBCHANNEL FLOW AREA	1.019	1.019	1.000	1.000	1.000	1.000
FILM COEFFICIENT	1.000	1.000	1.120	1.120	1.120	1.120
PELLET ECCENTRICITY	1.000	1.000	1.174	1.000	1.000	1.000
CLADDING THICKNESS & CONDUCTIVITY	1.000	1.000	1.000	1.000	1.000	1.000
SODIUM PROPERTIES	1.017	1.017	1.000	1.000	1.000	1.000
FLOW DISTRIBUTION CALCULATION	1.060	1.060	1.005	1.005	1.005	1.005
NUCLEAR DATA	1.070	1.070	1.070	1.070	1.070	1.070
CRITICALITY	1.010	1.010	1.010	1.010	1.010	1.010
FISSILE DISTRIBUTION	1.030	1.030	1.030	1.030	1.030	1.030
TOTAL DIRECT UNCERTAINTY	1.125	1.125	2.334	2.047	1.078	1.078
TOTAL STATISTICAL UNCERTAINTY						
+3 SIGMA	1.117	1.117	1.225	1.143	1.143	1.143
+2 SIGMA	1.078	1.078	1.150	1.095	1.095	1.095
TOTAL +2 SIGMA UNCERTAINTY (Temperature Rise Multipliers)	1.213	1.213	2.684	2.242	1.180	1.180

	CLADDING TEMP RISE FOR CLAD TEMP		CLADDING TEMP RISE FOR FUEL TEMP		FUEL GAP TEMP RISE FOR FUEL TEMP		FUEL TEMP RISE FOR FUEL TEMP	
	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL
DIRECT UNCERTAINTIES								
POWER LEVEL AND CONTROL DEADBAND	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030
INLET FLOW DISTRIBUTION	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
FLOW CALCULATION	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
CLADDING CIRC VARIATION	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
PELLET ECCENTRICITY	1.014	1.000	1.000	1.000	1.000	1.000	1.000	1.000
PHYSICS MODEL AND ROD BANK	1.040	1.040	1.040	1.040	1.040	1.040	1.040	1.040
STATISTICAL UNCERTAINTIES								
INLET FLOW DISTRIBUTION	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
WIRE WRAP ORIENTATION	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
SUBCHANNEL FLOW AREA	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
FILM COEFFICIENT	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
PELLET ECCENTRICITY	1.174	1.000	1.000	1.000	1.000	1.000	1.000	1.000
CLADDING THICKNESS & CONDUCTIVITY	1.050	1.050	1.050	1.050	1.000	1.000	1.000	1.000
SODIUM PROPERTIES	1.000	1.000	1.000	1.000	1.017	1.017	1.000	1.000
FLOW DISTRIBUTION CALCULATION	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
NUCLEAR DATA	1.070	1.070	1.070	1.070	1.070	1.070	1.070	1.070
CRITICALITY	1.010	1.010	1.010	1.010	1.010	1.010	1.010	1.010
FISSILE DISTRIBUTION	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030
TOTAL DIRECT UNCERTAINTY	1.086	1.071	1.071	1.071	1.071	1.071	1.071	1.071
TOTAL STATISTICAL UNCERTAINTY								
+3 SIGMA	1.197	1.092	1.092	1.092	1.079	1.079	1.077	1.077
+2 SIGMA	1.131	1.061	1.061	1.061	1.052	1.052	1.051	1.051
TOTAL +2 SIGMA UNCERTAINTY (Temperature Rise Multipliers)	1.229	1.137	1.137	1.137	1.127	1.127	1.126	1.126

COOLANT RADIAL (INTRA-ASSEMBLY) PEAKING FACTOR (Temperature Rise Multiplier) 1.12

COOLANT INLET TEMPERATURE UNCERTAINTY (F) (Add To Inlet Temperature) 4.9

	ASSEMBLY OUTLET TEMPERATURE		GAS PLENUM TEMPERATURE		BURNUP
	BOL	EOL	BOL	EOL	
DIRECT UNCERTAINTIES					
POWER LEVEL AND CONTROL DEADBAND	1.030	1.030	1.030	1.030	1.020
INLET FLOW DISTRIBUTION	1.020	1.020	1.020	1.020	1.000
FLOW CALCULATION	1.030	1.000	1.030	1.030	1.000
CLADDING CIRC VARIATION	1.000	1.000	1.000	1.000	1.000
PELLET ECCENTRICITY	1.000	1.000	1.000	1.000	1.000
PHYSICS MODEL AND ROD BANK	1.030	1.030	1.040	1.040	1.040
STATISTICAL UNCERTAINTIES					
INLET FLOW DISTRIBUTION	1.059	1.059	1.059	1.059	1.000
WIRE WRAP ORIENTATION	1.000	1.000	1.010	1.010	1.000
SUBCHANNEL FLOW AREA	1.000	1.000	1.019	1.019	1.000
FILM COEFFICIENT	1.000	1.000	1.000	1.000	1.000
PELLET ECCENTRICITY	1.000	1.000	1.000	1.000	1.000
CLADDING THICKNESS & CONDUCTIVITY	1.000	1.000	1.000	1.000	1.000
SODIUM PROPERTIES	1.017	1.017	1.017	1.017	1.000
FLOW DISTRIBUTION CALCULATION	1.000	1.000	1.060	1.060	1.000
NUCLEAR DATA	1.070	1.070	1.070	1.070	1.070
CRITICALITY	1.010	1.010	1.010	1.010	1.010
FISSILE DISTRIBUTION	1.030	1.030	1.030	1.030	1.030
TOTAL DIRECT UNCERTAINTY	1.082	1.082	1.125	1.125	1.061
TOTAL STATISTICAL UNCERTAINTY					
+3 SIGMA	1.098	1.098	1.117	1.117	1.077
+2 SIGMA	1.066	1.066	1.078	1.078	1.051
TOTAL +2 SIGMA UNCERTAINTY (Temperature Rise Multipliers)	1.153	1.153	1.213	1.213	1.115

4.24 Comment

A description of how these uncertainties (assumed to be released to Comment 4.23) were factored into the PRISM safety analysis and the resulting conclusions regarding resulting off-site doses.

Response

As discussed in Section 4.3.2.5 the control rods compensate for reactivity uncertainties associated with

Burnup
Temperature Defect
Criticality Prediction
Fuel Enrichment
Fuel Management

These uncertainties, together with a reactivity margin, and the reactivity due to burnup swing and temperature defect were used to determine the shutdown reactivity requirements. Each control rod assembly has been designed with sufficient worth to shutdown the reactor from hot, full power condition to cold, zero power condition even if the remaining five rods were withdrawn to the normal full power operating position.

As Table 4.3-10 shows, a single control assembly containing naturally-enriched boron carbide is capable of reactor shutdown from hot full power to cold shutdown conditions with a shutdown margin of .74$ at beginning of equilibrium cycle (BOEC) and a margin of .79$ at end of equilibrium cycle (EOEC).

The thermal and hydraulic uncertainties are discussed in Section 4.4.

Two-Sigma values have been estimated for the following parameters:

Outlet temperatures at beginning and end of life (BOL and EOL) for each fuel assembly. (Fig. 4.4-4)

Peak cladding midwall temperature at BOL and EOL for each assembly. (Fig. 4.4-6)
Upper bound (two-sigma) BOL fuel assembly outlet temperatures and lower bound (-two-sigma) blanket and control assembly outlet temperatures for use in estimating the maximum potential for thermal striping on above core structures. (Fig. 4.4-7).

As concluded in Section 4.4, the temperature environment for the core is mild and is expected to produce very little cladding life degradation. The temperature difference between fuel and internal blanket discharge coolant indicates the need to carefully define the refueling batch patterns to avoid placing fresh fuel adjacent to fresh blanket assemblies. Such detailed design and analysis have not yet been performed.

Uncertainties in the core response to the reactivity insertion DBE of uncontrolled rod withdrawal at 100% power is discussed in Section 15.4.1.

4.24 Response (Continued)

Hot channel factors have been used to estimate two-sigma values for the core outlet temperature, the peak fuel temperature, and the peak cladding temperature. Table 15.4-1 presents the hot channel factors used in the ARIES-P analysis.

Factors considered include:

Sodium Temperature Rise from Inlet,
Sodium Inlet Temperature,
Film Temperature Drop,
Cladding Temperature Drop,
Fuel Surface to Inner Node Temperature,
Fuel Cladding Gap Temperature,
Intra-Assembly Sodium Radial Peaking,
Core Wide Outlet Temperature Increase.

Section 15.4.1.2, p. 15.4-1 describes how uncertainties were combined.

The results presented in Section 15.4.1.3 show that the peak temperatures remain well below the design limits for unlikely events, with allowance for uncertainties at the two-sigma level. (Fig. 15.4-3 and Fig. 15.4-4).

Uncertainties in the primary coolant and vessel temperatures due to loss of normal shutdown cooling DBE are discussed in Section 15.5-1.

Table 15.5-1 presents expected and conservative parameters for evaluation. The used conservative parameter values are believed to result in two-sigma level of reactor temperatures. These parameters include:

Decay Heat,
Heat Transfer Coefficients,
Thermal Emissivity,
Bottom Head Heat Loss.

Section 15.5.1.3 presents the results of analysis which show that the reactor vessel temperature remains below the design limit even for the conservative case analyzed.

Section 15.6 presents test and analysis results which confirm local fault tolerance of the metal fuel used in PRISM.

Section 15.6.4.1 provides reasons as to why increased heat generation by enrichment error or oversized fuel is extremely remote and why the metal fuel will tolerate such errors without failure. Past experience with metal and oxide fuels are used in the discussion but no quantitative analysis is provided at this time. (See response to Comment 4.10)

Section 15.6.4.2 provides reasons as to why blockages and bond defects are very unlikely and how the metal fuel will tolerate such defects, as has been verified with TREAT and EBR-II experiments. (See response to Comments 4.5 and 4.21.

4.24 Response

Section 15.6.5 presents evidence on the capability of metal fuel to accommodate fuel failure without propagation. (See response to Comment 4.1)

No design basis accidents have been identified that have significant radiological releases to containment.

As discussed above, the PRISM core has been designed with substantial shutdown and thermal hydraulic margins. These margins, and the PRISM inherent safety, prevent fuel failure even under extremely unlikely beyond design basis events. Consequently, only random failure of two fuel pins accompanied by the extremely unlikely spill of 1000 gallons of Na in the cold trap, or accompanied by an extremely unlikely leak of cover gas, were considered as design basis accidents for evaluation of site doses (Section 15.7 and 15.9). Failure of five fuel pins during refueling was also analyzed in Section 15.8. As shown in Section 15, these accidents lead to site doses which are substantially less than the site dose criteria of 10CRF100.

4.25 Comment

An assessment of the sensitivity of the PRISM safety analysis to variations in these uncertainties, such that an assessment can be made of the margin between the estimated uncertainties and those which the PRISM can accommodate and still maintain radioactive releases within the limits of the top level criteria.

Response

The radiological release to the reactor cover gas for the design basis accidents and for a significant fraction of beyond design basis accident is very small (fission gas and vapor fractions associated with the radiological inventory in several fuel pins). As discussed in Chapter 6.0 and the PRA the containment can accommodate a large release of fission products for which there is no credible mechanism. This release was selected to provide defense in depth and assure large margins in the containment capability.

4.26 Comment

How is your fuel different from EBR-II fuel? Is the existing data base sufficient to demonstrate:

a - extended burnup capability?
b - transient fuel performance?
c - fission product behavior?
d - What data are you using as your base?

4.26 Response

The fuel differs in two major areas. (1) The PRISM design calls for 26 w/o Pu in the fuel. EBR-II irradiations have focused on fuel with Pu content of 20 w/o and below. One test assembly, X432C, contains pins with 26 w/o Pu. This assembly has reached 3.5 at % burnup. Argonne is increasing the range of Pu content being used in the metal fuel irradiation program. (2) PRISM design calls for a fuel column length of 47 inches while the EBR-II core length is 13 1/2 inches.

Is the existing data base sufficient to demonstrate:

a) Extended burnup capability - Current irradations have not reached sufficiently high burnups (beyond 14-15%) to determine when cladding loading, from either high plenum pressures or from fuel-cladding mechanical interaction caused by the exhaustion of the fuel porosity by solid fission products is sufficient to fail the cladding.

b) Transient fuel performance - Insufficient tests have been performed to characterize metal fuel performance over a wide range of transient behavior. Before commercialization of metal fuel can occur, additional transient testing will be required. However, early results from TREAT tests Ms, M3, M4, M5 and M6 indicate that the factors that govern the transient behavior of metallic fuel are relatively simple, and well understood.

c) Fission product behavior - Most fuel swelling is caused by gaseous fission products. A considerable quantity of data from EBR-II Mark-II driver pins and from recent U-Pu-Zr irradiations has shown that metallic fuel designs with 75% smear density develop enough interconnected porosity so that after several atom percent burnup fissions gases are released to the plenum at the same rate they are being generated. Although some gas remains in the fuel, the PRISM plenum volume is sized to accommodate 100% fission gas release, thus alleviating the need to predict fission gas release rates accurately.

   Solid and liquid fission product swelling may close off some of the interconnected porosity late in life, depending on how much of the liquid products such as cesium are carried to the plenum by the bond sodium. Irradiation tests being run to cladding breach will indicate whether or not solid fission product swelling contributes to cladding breach.

d) What data are you using as your base? - The current data base for steady-state and transient performance are given in the attached Tables I and II. Calculations used to analyze these tests have used properties taken from the, "Metallic Fuels Handbook," compiled by G. L. Hofman, L. Leibowitz, J. M. Kramer, M. C. Billone and J. F. Koenig (ANL-IFR-29).

Table I. Metallic Fuel Irradiation Status, May 1987.

EXPERIMENT	Pu CONTENT	CLADDING	PIN O.D.	CURRENT BURNUP,at.%	STATUS
EBR-II X419B	0, 8, 19	D9	0.230	8.9	PIE at 1,2,3 at.%; RTCB
EBR-II X420A	0, 8, 19	D9	0.230	9.3	PIE at 6 at.%; RTCB
EBR-II X421	0, 8, 19	D9	0.230	10.0	PIE at 10 at.%; RTCB
EBR-II X423C	0,3,8,19,22,26	316	0.290	3.5	PIE at 0.5,0.9,2,(5) at.%
EBR-II X425A	0, 8, 19	HT9	0.230	5.2	PIE at 3,(6) at.%; RTCB
EBR-II X428	8, 19	316	0.174	2.5	Complete
EBR-II X429	0, 8, 19	HT9/316	0.230	2.4	PIE at (8) at.%; RTCB
EBR-II XY24	19	316	0.174	2.5+ 40 days	Goal 2.5+ ~140 days
EBR-II XY27	8	316	0.174	2.5+ 0 days	Goal 2.5+ ~256 days
EBR-II X430	0, 19	HT9	0.290	0	Starts in May
FFTF IFR-1	0, 8, 19	D9	0.270	2.8	Goal of 10 at.%
FFTF MFF1&1A	0	HT9	0.270	0	Starts in June

Table II. Metallic Fuel Transient Test Status, May 1987.

Experiment	Fuel	Cladding	Burnup, a/o	Test Type
M1	U-Fs (segments)	SA316	3.5	TOP in dry capsule
M2	U-Fs	SA316	0.35, 4.4, 7.9	TOP in flowing sodium
M3	U-Fs	SA316	0.35, 4.4, 7.9	" " " "
M4	U-Fs	SA316	0.0, 4.4, 2.4	" " " "
M5	U-19Pu-10Zr	D9	0.8, 1.9	" " " "
M6	U-19Pu-10Zr	D9	1.9, 5.3	" " " "

4.27 Comment

Will the FFTF metal core be more prototypical of your reactor design?

Response

Satisfactory behavior must be demonstrated for higher Pu content fuels. Current PRISM design calls for fuel with 26 w/o Pu. With the exception of one experiment (X423C) containing some pins with 26 w/o Pu, which has completed 3.5 at % burnup, EBR-II irradiations have investigated Pu contents of 20 w/o and below. ANL is increasing the range of Pu content being used in the EBR-II test program.

The FFTF Series III metal driver fuel will be more prototypical of the PRISM design with respect to fuel column length, peak-to-average power and fluence-to-burnup ratio. However, the Series III driver fuel is a U-10 w/o Zr alloy while the reference PRISM fuel is U-26 w/o Pu-10 w/o Zr. The most important factors governing steady-state and transient fuel performance depend on local conditions at a given axial elevation (e.g., alloy content; smear density; radial porosity and fission product distributions; and radial fuel and cladding temperature distributions). The effects of geometric factors such as fuel-to-plenum volume can be determined by scaling data from irradiations of shorter EBR-II ternary fuel pins. Comparison of EBR-II data on U-10 w/o Zr with FFTF data on U-10 w/o Zr will demonstrate the validity of models used in this scaling procedure. In addition, experimental FFTF subassemblies, such as IFR-1, will contain some full-length pins with a range of Pu content. These tests will provide a direct comparison between the irradiation performance of EBR-II pins and full-length pins with prototypic alloy compositions.

4.28 Comment

Referring to Item 8 of the operational requirements on P 4.2-4, why isn't the core reactivity feedback negative for entire power range (0-100%)?

- What is the breakdown for the feedbacks?

Response

The PRISM core restraint is similar to that used in CRBRP. The core restraint system holds the assemblies from the perimeter and does not have a core support grid at both the top and bottom of the core. This type of restraint system must have gaps between the assemblies when at shutdown temperature to allow assembly replacement. These gaps between restraint locations, or load pads, are made as small as reasonably possible, so that the restraint plane remains as tight as possible and still allows refueling. Upon rise to power, the thermal gradients generated across the core bow the assemblies. Until the load pads bind up against a perimeter boundary, called a former ring, or against each other, as in compaction toward the core centerline the assemblies are theoretically free to move a small amount. Until bound

4.28 Response

up, the core restraint system, theoretically at least, is in an indeterminate state and the potential exists for an assembly to change its direction of lean and create a reactivity change. Such a change would be very small relative to the overall reactivity coefficient effects; however it would appear as a small sudden positive or negative feedback. For this reason, the core instantaneous reactivity feedback coefficient does not become assuredly negative until a power level of about 40% to 60%, when the thermal gradients have become large enough to assure a bound state within the core restraint system.

The real core has friction between assemblies and inelastic bow in the partially burned assemblies and thus tends to have a partially bound up restraint system. These effects then cause the bowing and total reactivity feedback to be negative. However, initially in an LMR restraint system, the overall gaps in the restraint planes, combined with the general bow shape, convex to the core centerline, means a slight core compaction must occur during the thermal gradient buildup. Thus, the restraint system will contribute a small positive feedback during the initial low power portion of the rise to power.

The attached table of reactivity feedback contributions for a quasi-steady state calculation of the rise to power (and beyond) indicates the typical relative magnitudes of the feedback mechanisms. In this case, an ideal core starting from ideal straight assemblies, the bowing of the core does not lead to a mathematically determinant geometry until a power level between 40% and 60% of full power is reached. Below that power level, the core restraint and geometry are predictable only and thus an initial feedback contribution for incremental power changes is unpredictable. As shown by the relative magnitudes of the feedback mechanisms, however, this indeterminancy is not actually a problem because bowing is such a small potential feedback compared to Doppler and fuel axial expansion feedbacks that the overall, net feedback remains negative.

4.28 <u>Response</u> (Continued)

TYPICAL REACTIVITY FEEDBACKS AS A FUNCTION OF CORE POWER LEVEL

CORE POWER % & (P/F)	CORE FEEDBACK (¢)						
	TOTAL	DOPPLER	FUEL AXIAL EXPANSION	CLADDING & DUCT DENSITY	SODIUM DENSITY	GRID PLATE EXPANSION	BOWING**
0	0	0	0	0	0	0	$0+A\pm B_0$
20	-28	-12	-15	0	6	-7	$-2+A\pm B_{20}$
40	-52	-24	-29	0	14	-14	$-4+A\pm B_{40}$
60	-81	-35	-44	0	21	-22	$-7+A\pm B_{60}$
80	-100	-44	-59	1	29	-29	-9+A
100	-123	-54	-73	1	36	-37*	-11+A
130	-143	-60	-88	1	40	-37*	-13+A
150	-157	-64	-97	1	42	-37*	-15+A
180	-177	-70	-111	1	46	-37*	-17+A
200	-190	-74	-120	1	48	-37*	-20+A

**A is a fixed magnitude offset that represents the reactivity worth of the pre-bowed condition of the core, where $-6¢ \leq A \leq +6¢$

B_i is a variable magnitude uncertainty that represents the feedback from one or more assemblies potential wobble within their nominal "hole" in the core restraint system. The magnitude depends on the bowed and prebowed core states. In general, $B_{60} \sim Ø$ ¢ and the fully bound restraint state for higher power levels (power to flow ratios) makes B_{60} and the higher B_i's also Ø. $B_Ø$, assuming all assemblies could "fall" from fully out to fully in, and vice versa, has a magnitude of approximately $\pm(11-|A|)$¢.

4.29 Comment

What are the fuel design methods and data base? Are there any validations? What are the uncertainties? Were these uncertainties taken into account in the fuel design? How would they impact the inherent safety?

Response

Estimated uncertainties in the neutronic analysis, thermohydraulic analysis, fuel fabrication, core operation and instrumentation are included in thermal and power "Hot Channel Factors." The $+2\sigma$ (of an approximately normal distribution) uncertainty factors are applied in design basis analyses in conceptual design. The factors used are based on the Clinch River Breeder Reactor hot channel factors as modified during post CRBRP/LMR studies. See the Table of Uncertainties in the response to Comment 4.23.

Inherent safety analyses currently use nominal (best engineering estimates) analyses without hot channel factors. Large margins between predicted performances and damage limits are being maintained for BDBAs so that planned full scale safety tests can be conducted. Such margins will allow confidence in the success of the safety test and its bootstrapped test series of successively more severe BDBAs.

4.30 Comment

Was the temperature defect uncertainty of $\pm 30\%$ applied to all temperature induced coefficients? The core radial expansion reactivity feedback due to thermal bowing and thermal expansion is expected to have a larger uncertainty (perhaps $\pm 50\%$).

- What reactivity effect is introduced from permanent channel deformation from void swelling in the ducts?

Response

The temperature defect of $\pm 30\%$ is based on individual uncertainties for each of the elements in the defect. Those that are applicable to the nuclear feedbacks have not been applied in transient analyses in any form other than the hot channel factors discussed in the response to Comment 4.29. These are applicable for design basis events. For beyond design basis events, sensitivity studies of the effects of uncertainties have only been performed on previous core designs.

For discussion of bowing feedback, see the responses to questions 4.6 and 4.28.

The void swelling in HT-9 is negligible.

4.31 Comment

The criticality prediction uncertainty of ±1$ seems too optimistic. This is equivalent to $k_{eff} = 1.0 \pm 0.005$. Please explain your basis for this uncertainty.

Response

The comment is in error; the uncertainty in k_{eff} is ±0.0031, not ±0.005. The latter value is the LWR requirement.

The reactivity criticality prediction is an assumed value for conceptual design. Later more detailed analyses may cause a change to this value. The liquid metal PRISM reactor has a more closely coupled core, due to its fast spectrum than light water reactors; therefore, the PRISM nuclear analysis is inherently more accurate and would be expected to have a reduced uncertainty in k_{eff}.

4.32 Comment

How much assembly deformation occurs before the IVTM will not operate on the assembly?

Response

The principal type of assembly deformation which can affect assembly withdrawal by the IVTM is bowing. However, bowing is restrained at the above-core load pad (ACLP) and top load pad (TLP) planes by the load pads on the surrounding assemblies and, at the TLP, by the fixed restraint ring just outboard of the core. The gaps between assemblies and between the outer ring of assemblies and the restraint ring are sized such that the maximum displacement (e.g., due to bowing) of the top of any assembly is less than the margin designed into the IVTM. Therefore, while assembly bowing can cause significant frictional withdrawal loads, it cannot prevent the IVTM operating on any assembly.

The In-vessel Transfer Machine (IVTM) transfers core assemblies within the reactor to the core, the fuel storage location, and the transfer station. The IVTM is designed to generate withdrawal and insertion loads of 3000 lbs. This load compares favorably with the design load capacity of the refueling machines of CRBRP (4000 lbs. pull, 3000 lbs. push) and FFTF (4000 lbs. pull, 3000 lbs. push).

In the event that the IVTM is unable to withdraw a core assembly in the normal operating mode, several options are available that will allow completion of the operation. Reducing the withdrawal speed to 2% of the rated speed will enable the assembly to be slowly "walked out" of the core. If the preceding operation is unsuccessful, then removing the assemblies surrounding the stuck assembly will loosen the core enough to allow its removal. Both of these options have been successfully used at FFTF.

4.32 Response (Continued)

A procedure will also be incorporated into the refueling schedule that will reduce the possibility of the core becoming excessively tight. This procedure is successfully used by FFTF and other operating breeder facilities. During every refueling various core assemblies will be withdrawn, rotated 180° and reinserted in the same location. This reorientation of the core assemblies will reduce the overall tightening of the core.

4.33 Comment

How will bowing be demonstrated to be predictable and always cause a core expansion?

- What is the bow's reactivity feedback change over the lifetime of the duct?
- What R&D will guarantee this?
- Could a control rod duct be displaced by a neighboring duct that has bowed? (Especially the control rod channels.)

Response

Restraint of the free-bowing of the core assemblies by the core restraint system generates negative bowing reactivity feedback; this is discussed in the response to Comment 4.6. A breakdown of the total reactivity feedback, showing the relative importance of the bowing feedback, is given in the response to Comment 4.28.

Irradiation swelling and creep of the assembly ducts can change the static shape of the ducts and, thus, the duct shapes when restrained by the core restraint system. This change in shape implies a change in bowing reactivity feedback. HT9 has been selected as the duct material because of its low irradiation swelling. In addition, the PRISM coolant temperatures are relatively low (875°F outlet) which also minimizes thermal creep.

Sufficient gaps between adjacent ducts have been provided to prevent contact of maximally bowed assemblies and duct-duct interaction. Changes in bowing reactivity feedback over a duct lifetime are therefore calculated to be very small.

The key related R&D testing is inspection of the full-length HT9 fuel assemblies irradiated in FFTF and determination of their swelling and creep as a function of fluence and temperature. Confirmatory information on bowing feedbacks and their change with fluence will be provided by the PRISM Safety Test.

The provision of sufficient gap between adjacent ducts to prevent contact of bowed assemblies also includes the displacement of a control rod duct by a bowed neighboring duct.

4.34 Comment

Is there a systematic error (constant bias) resulting from the design methods in addition to the random errors considered? Are these sources of error treated separately?

4.34 Response

The neutron energy group number and structure, the cross section data base choice (ENDF-BV vs. other), and diffusion vs. transport calculations lead to systematic uncertainties (or errors) in the nuclear analyses. These are included in the power uncertainty in the Hot Channel Factors as Direct Uncertainties and are not combined as statistically independent uncertainties. (See the Table of Uncertainties in the response to Comment 4.23)

4.35 Comment

For logarithmic representation of Doppler reactivity, the Doppler constant α_D should be defined as

$$\alpha_D = T^{3/2} d\rho/dT$$

for the metal fuel. Was the reported a_D calculated as above? Note that the temperature slope should be that of the reactivity ρ, not the effective multiplication factor k. Was the reported α_D computed with dk/dT?

Response

The Doppler coefficient is defined as

$$\alpha_D = T^n \, d\rho/dT$$

For a thermal neutron spectrum, low energy resonances cause n to be about 1.5. For a fast neutron spectrum, the low energy resonances become less important. For an oxide-fueled LMR, n ~ 1.05 to 1.1. (See ZPR-TM-394, Reactivity Coefficients, April 1983, ANL).

While it is true that the value of n is theoretically expected to increase as the neutron spectrum hardens, as it does in going from an oxide to a metal core, actual measured data from the metal fueled core benchmark critical assembly, ZPPR-15 reaffirms the status for oxide cores, that the value of n cannot be experimentally distinghuished from unity.

4.35 Response (Continued)

From ZPPR Experiments:

Assembly	Core Type	Sample Type	n: E*	n: C
15A	U/Pu/Metal	DU Metal	0.8	
15B	U/Pu/Zr Metal	DU Metal	1.1	
15B	U/Pu/Zr Metal	UZr Metal	0.9	1.07
15D	U/Zr Metal	DU Metal	1.06	
15D	U/Zr Metal	UZr Metal	1.02	1.08
11B	Oxide CRBR	Oxide	0.98	
11B	Oxide (BOC)	Homo. Fuel		1.01**
11C	Oxide (EOC)	Oxide	1.01	1.11**
11F	Voided Oxide	Oxide	0.96	

* 1σ - 10%
**Version IV data

4.36 Comment

Is there any burnup induced effect which might limit the axial fuel expansion?

Response

Prior to approximately 3 at% burnup, the fuel is not in contact with the cladding and will expand based on its own temperature and thermal expansion coefficient (including the effects of trapped gas). Due to fuel swelling, after about 3 at% burnup, the fuel is in contact with, and bound by the cladding and will expand less. The lower limit on the reduced fuel axial expansion is the cladding axial expansion.

4.37 Comment

What is the total density reactivity coefficient on Table 4.3-2 of Amendment 1 that is strongly positive?

Response

The total density coefficient of Table 4.3-2 is the reactivity effect of an increase in the atom densities in the fixed size core cells.

$$\frac{\rho dk}{d\rho} = \frac{K(\text{Normal Atom Density}) - K(\text{Perturbed Density})}{(\text{Fractional Density Change})}$$

4.37 Response (Continued)

It is strongly positive because an increase in the core atom density improves neutronic efficiency in fuel fission captures more than degrading it with additional parasitic captures.

4.38 Comment

On Table 4.3-9 of Amendment 1, the burnup reactivity swing has been increased from 0.37$ to 0.65$. Is this the result of a new design or a new definition of terms?

- This almost seems to be within the uncertainty of criticality predictions.

Response

Table 4.3-9 of Amendment 1 actually shows 0.06$ rather than 0.65$ for the burnup reactivity swing. The 0.65$ is either a typo or a misreading of the table. See also the table provided in the response to Comment 4.39. The reduction in burnup reactivity swing from 0.37$ to 0.06$ is the result of a new design. The core height has been increased one inch to increase the internal conversion. The core is specifically designed to have a low burnup swing. Sufficient excess reactivity will be designed into the core to cover uncertainties, however, the core design and reloads will specifically be designed to yield the minimum excess reactivity.

4.39 Comment

On Table 4.3-11 of Amendment 1, the burnup reactivity swing and uncertainty of 0.06$ does not seem to include the uncertainty. What is the uncertainty ?

- What is the excess reactivity required?

Response

The 0.06$ is not an uncertainty, but is the reactivity swing to be added to the uncertainty that a single rod scram should offset. The attached Table makes this clearer.

4.39 Comment (Continued)

CONTROL REACTIVITY COMPONENTS ($)

COMPONENT	SYSTEM WORTH REQUIREMENT	Nth CORE OPERATIONAL UNCERTAINTIY[3]	REACTIVITY SUPPRESSED AT FULL POWER
TEMPERATURE DEFECT[1]	1.43		
BURNUP REACTIVITY SWING	0.06		0.06
SHUTDOWN MARGIN	1.00		
UNCERTAINTIES			
Temperature Defect	0.29	0.15	
Burnup Reactivity	0.01	0.01	
Criticality Prediction	1.00	0.10	
Fissile Loading	1.00	0.20	
Refueling	1.00	0.10	
Total[2]	1.76	0.29	0.43
TOTAL	4.25		0.49

[1] Cold Shutdown (400°F) to full power.

[2] Statistically combined as independent and random.

[3] Assumes a "known" core with calibrated analysis codes and batch refueling.

4.40 Comment

Was the direct heating of core structures and core support structures by γ rays and fast neutrons taken into account in the calculation of thermal expansion and thermal bowing?

- Gamma heating might reduce the predicted worth of channel bowing.

Response

The power distribution include γ captures and fast neutron but not γ transport heating within core structures. Fuel and blanket structure temperatures without γ transport heating are adequate for conceptual design. The power density of the shield assemblies including γ transport heating has not been calculated.

The reactivity feedback error induced by γ heating will not be large because most of the bowing feedback comes from the distortion of the outer ring of fuel. The temperature distributions of the fuel and blankets (internal and radial) are not greatly affected by the γ transport heating. About 95% of their power comes from neutron fissions.

The gamma and neutron internal heat generation within the core support structure has not been accounted for in the conceptual design. The heating (or cooling) is due entirely to coolant temperature changes.

4.41 Comment

Does the listed core radial expansion include thermal bowing? If so, how were the thermal bowing and thermal expansion combined? What are the influence of swelling and creep on radial expansion as burnup increases? What are the plans to verify the methodology used to calculate core deformation including radial expansion?

Response

The radial feedback coefficient accounts for all sources of radial expansion and assumes uniform radial expansions at all elevations equal to the center plane expansion.

Because core geometry is strongly dependent upon the bowing distortion and the restraint position movements (through cumulative gaps), the core geometry must be solved simultaneously considering top former ring expansion, assembly load pad expansions in each ring, grid plate expansion, assembly inelastic distortion and assembly elastic bowing. Either 2-D transient or 3-D quasi static solution methods are used. These detailed results for a limited set of environmental conditions are simplified into a correlation for radial feedback as a function of power to flow ratio or as a function of the change in core temperature rise from inlet to outlet [$\Delta(\Delta T)$] for the system-level transient analysis codes.

Also see response to Comment 4.28.

4.42 Comment

Could the channel ducts round out from elastic deformation and increase peaking factors which might induce voiding?

Response

Elastic deformation can, to some degree, round out the ducts and increase peaking factors; this effect has been included in the calculations of the peaking factors.

4.43 Comment

Thermal maldistribution could cause a misalignment with the UIS and the control rods. How is this avoided?

Response

Thermal maldistribution could cause the UIS structure to bow and cause misalignment between the control rod drive lines and the control assemblies. Recent thermal hydraulic analysis using the COMMIX code has shown that thermal mixing in the region above the core is very good and the sodium temperature and flow distribution in the annular gap between the UIS support cylinder and the core barrel extension structure are very uniform. Also, the design can accommodate and function with the considerable UIS deflections that occur during seismic events. Therefore, we cannot see any problems with any minor thermal distortions.

4.44 Comment

What is the design basis for orificing?

Response

There are three criteria which the core orificing must satisfy. (1) The orificing is to provide a uniform +2 sigma beginning-of-life peak cladding midwall temperature of 1100°F (or less) for all fuel assemblies and a uniform +2 sigma end-of-cycle peak cladding midwall temperature of 1150°F (or less) for all blanket assemblies. (2) The difference in ±2 sigma coolant outlet temperatures of adjacent assemblies (e.g., new fuel assembly near new blanket, new fuel near control, spent blanket near control, etc.) must be less than 370°F to limit the thermal striping potential. (3) To prevent thermal aging of the upper internal structure, the maximum assembly outlet temperature is limited to 1130°F.

4.45 Comment

How fast can the shutdown system insert control rods (from the time a parameter exceeds its trip value until rods begin to move? Until they are fully inserted?)

Response

The gravity drop scram achieves full travel within 2 seconds, the back-up drive-in scram requires 18 seconds for full travel.

See response to Comment 4.8.

4.46 Comment

There is no separate redundant shutdown system designed. How do you demonstrate shutdown redundancy? With small (0.93$) shutdown margin, it is not clear that any one of six control rods can provide cold shutdown when the uncertainties are included.

Response

Shutdown redundancy is provided in two ways. First, any one of the six control rods alone can shutdown the reactor. Second, there are, within each control rod, two independent methods of scramming: latch release and gravity drop, and powered drive-in.

A 0.93$ margin exists for a one-rod shutdown. Any one of the six rods provides cold shutdown because all temperature defects are included in the reactivity calculations. The Δk effect by a one-rod scram includes the summation of temperature defects (-1.40$), the summation of Δk of the operating core (-0.29$) and the total burnup Δk for the scramming rod (-0.01$). The other five rods are assumed to suppress their burnup reactivity allocation in the full-power banked position.

4.47 Comment

How are the rigid and flexible absorber bundles arranged in the absorber channel? Provide an illustration to show their relationship.

- Where are they positioned in the core?

Response

Since preparation of the PSID Chapter 4, the design has been simplified to include only one type of absorber bundle. Studies are continuing to select the preferred design. The seleted design, either rigid or flexible, will be used in each of the six control assembly core positions.

4.48 Comment

Could the fingers on the latch be calibrated to expand enough on high temperatures to release the absorber rod to give this system an inherent hot temperature SCRAM?

Response

Yes, at an early stage of the design, this concept was actually incorporated. However, it was determined that, at each startup from refueling, calibration for the local temperature environment would require manual adjustment of a fine motion screw. The concept as currently designed provides adequate diversity and reliability and it appeared desirable to avoid the complexity of operational adjustment.

Responses to NRC Comments on the ANL Metal Fuel Program

ANL presented their Metal Fuel Program for LMR's to the NRC reviewers on May 13, 1987. NRC developed seven comments as a result of this meeting. These comments were generic to LMR's, and ANL and HEDL provided GE with generic responses to these comments. We have used this information in preparing the following responses, 4.ANL-1 through 4.ANL-7. See Page FA-50 for additional discussion on the Metal Fuel Program.

4.ANL 1

Please provide a description of the transient TREAT and "run beyond cladding breach" testing programs planned in support of the metal fuel design.

Response

The TREAT program in support of the IFR and the Run Beyond Cladding Breach (RBCB) programs are summarized by ANL in the following paragraphs. GE notes that TREAT testing is currently limited to a maximum Pu content of 19 wt%, while PRISM uses U-26.4Pu-10Zr ternary fuel. In general, the transient overpower and breached behavior of ternary fuel are very similar and one can extrapolate from test results on U-19Pu-10Zr fuel to U-26.4Pu-10Zr fuel. However, GE is recommending the use of prototypic PRISM fuel in at least one future TREAT test.

A program of in-pile safety experiments is being conducted in TREAT on metal-alloy reactor fuel to study the response of the fuel to severe off-normal conditions. Information from the experiments is essential for the development and validation of models in LMR safety codes.

Six experiments have been performed so far in the program. The first, test M1 was performed in an inert-gas capsule to optically measure the elongation of short segments of fuel pins during transient heating through the fuel melting point. All subsequent tests, i.e., M2 through M6, have been conducted with whole fuel pins in flowing-sodium loops.

The loop tests were performed to obtain information on two key fuel behavior characteristics under transient overpower (TOP) conditions in metal-fueled reactors: the margin to cladding breach and the axial swelling of fuel within intact cladding. Tests M2, M3, and M4 were underway before irradiated IFR-type fuel was available; EBR-II driver fuel (U-5Fs in 316 SS cladding) was therefore used as a suitable substitute. Subsequent testing (M5 and M6) has been on the IFR fuel U-Pu-Zr. In each loop test, two U-Pu-Zr pins of three U-5Fs pins were located in separate flowtubes. The coolant flow rate was set independently for each pin by orificing. Fuel burnup was the principal parameter varied in these tests, since fuel swelling and cladding failure models have predicted a strong sensitivity to burnup. Some tests were intentionally continued until cladding failure occurred in order to observe the full amount of prefailure fuel elongation and to determine the conditions required to cause failure. Others tests were terminated shortly before failure in order to metallographically study the characteristics of the fuel and cladding at incipient cladding failure.

In these overpower simulations, steady nominal coolant flow rate was maintained during the test, except as perturbed by postfailure thermal-hydraulic events. Fuel was heated at the lowest rate of exponentially-increasing power consistent with the requirement of causing cladding failure within the energy limitations of a single TREAT transient.

Cladding failure in the tests was driven by (i) penetration of the cladding by eutectic formation between fuel and cladding and (ii) internal overpressure.

Peak power levels achieved at cladding failure were about four times normal power. This corresponds to a peak fuel-cladding interface temperature that is above a certain threshold temperature: the temperature at which the rate of fuel-cladding eutectic formation increases dramatically, resulting in penetration of the cladding within about one second. Out-of-pile work indicates that this temperature is about 1050-1080°C. The rate at which the power was increased was relevant to overpower accident scenarios for reactors with control rod worths initially considered for the IFR concept. Although diminished control rod worths in recent designs have led to lower power ramp rates, the rate used in the TREAT tests is sufficiently low that the fuel is heated quasi-statically. Therefore the tests are representative of the range of heating rates at which the fuel-cladding interface temperature rises within a few tens of seconds to the threshold for rapid eutectic penetration of the cladding. The observation that cladding failure in the tests performed to date has consistently occurred at approximately four times normal power confirms a central role played by that temperature threshold. It is expected that heating rates lower by an order of magnitude would result in cladding failure at a somewhat lower power, at temperatures where the mechanism for rapid eutectic attack does not act.

As the fuel burnup increases, the plenum pressure contributes more strongly to cladding failure, and less thinning of the cladding by eutectic is required. At high burnup, failure can occur before significant attack of the cladding takes place. It is importanct to note that, at the rate at which the fuel is heated in the tests, even a small amount of thinning requires that the temperature range where rapid eutectic penetration occurs be reached.

Fuel melting is calculated to begin at roughly three times normal power. A straightforward model of fuel swelling upon melting has been developed and validated on the basis of the data obtained from the initial tests performed on U-5Fs fuel, over nearly the entire range of burnup of interest to reactor operation. Fuel elongation of up to 20% was both predicted and measured. That peak value applies to a low burnup at which the pressure in small fission-gas bubbles in the fuel greatly overbalances the pressure of fission gas in the pin plenum. At high burnup, the elongation was much lower, about 3 to 5%. Validation of the model, or a modification thereof, remains to be accomplished for ternary fuel. Data from tests M5 and M6 on U-19Pu-10Zr fuel of burnups in the range 1 to 5 at.% have shown that ternary fuel swells much less upon melting than does the U-5Fs. The reason for this difference is yet unclear but is suspected to be a result of the higher temperatures to which the ternary fuel was subjected during its pre-irradiation and consequently a smaller amount of participating fission gas in the molted fuel. For the ternary fuel, it is tentatively concluded that fuel elongation will exceed a minimum value of 2 to 4% at all burnups greater than about 1 at.%, since that swelling can occur simply by expansion of fission gas in large pores (which occupy about one-fourth of the fuel volume) that are closed off upon fuel melting. Measurement of transient fuel elongation during the test was done using the TREAT fast neutron hodoscope. Verification of the final distribution of fuel was provided by posttest hodoscope scans, neutron radiography, and destructive examination of the fuel.

Posttest examination also revealed that thinning of the cladding by eutectic formation was slight in pins that did not fail but that much alloying of the fuel and cladding occurred in pins that did fail. This finding is consistent with a sharp temperature threshold for the rapid penetration associated with cladding failure in these tests.

In the five loop tests performed to date, failure was induced in three U-5Fs fuel pins and one U-Pu-Zr fuel pins. All failed at the extreme top of the fuel column, the location of highest cladding temperature and therefore the expected failure location in view of the failure mechanism involved: overpressurization of cladding weakened by elevated temperature and eutectic attack.

In all the pins that failed, a large fraction (40-80%) of the fuel was ejected from the cladding upon failure. The ex-pin fuel motion was nearly all upward, away from the fuel zone. The large distance that the fuel traveled in the coolant channel indicates a high degree of mobility, as would occur if the fuel had a melting point lower than the temperature of the outlet sodium at pin failure. Metallography of the fuel debris showed that the all of the expelled fuel had alloyed with steel from the cladding and/or flowtube. Over a wide range of composition, the fuel-steel alloy has a melting point that is indeed lower than the outlet sodium temperature when cladding failure occurred.

In all cases involving pin failure, the flow channel was never completely blocked by solidified masses of fuel or fuel-steel alloy; a coolable situation was maintained. This result is likely related to the high mobility of the fuel-steel alloy in the sodium flow.

Future experiments, approximately two per year, will be performed to provide additional evidence of fuel behavior under severe off-normal conditions as necessary for model development and validation to support metal-fueled reactor safety analyses. Because of current plans to convert the FFTF core to one of U-Zr fuel in HT-9 cladding, near-term experiments will focus on testing fuel pins of that type in order to provide support for safety assessments of the new core. Through the longer-term, the objective is to support SAFR and PRISM licensing in the early 1990's. Specific experiment goals and characteristics, e.g., burnup levels, overpower or undercooling conditions, amount of fuel and cladding damage at test terminations, use of EBR-II-length or full-length fuel, will be selected to most effectively meet the analytical needs as those needs evolve.

Ref. A. E. Wright, T. H. Bauer, R. K. Lo, W. R. Robinson, and R. G. Palm, "Recent Metal Fuel Safety Tests in TREAT," Proc. ANS/ENS Int'l. Conf. on the Science and Technol. of Fast Reactor Safety, Guernsey, England, May 12-16, 1986, CONF-860501-9, Vol. 1, p. 59.

4.ANL 2

At the meeting on May 13, 1987 we were told that safety analyses involving metal fuel assumed that 90% of the fission gas is released. If, instead, a 100% release occurred, how would this affect your safety analysis?

Response

All PRISM safety analyses assume 100% fission gas release. Studies by ANL indicate that this assumption may be conservative and a 90% release assumption may more closely approximate the actual behavior.

ANL estimates that the use of 100% gas release, rather than 90% release, results in decrease in rupture life by a factor of 0.8. This decrease is consistent with deformation-based failure results and is well within the normal scatter in rupture data.

4.ANL 3

Please discuss the sensitivity of fuel failure to uncertainties in clad temperature.

Response

o Sensitivity of Transient Cladding Failure to Fission Gas Release and Temperature

The metallic fuel design for the PRISM reactor incorporates sufficient porosity (75% smear density) to allow significant gas release once the fuel swells out to the cladding early in life. The amount of gas retained in the fuel after that time is approximately constant. Section 1 provides a simple model for determining the gas retention in the fuel pins and the related gas release and plenum pressure. These results are used in Section 2 to determine the sensitivity of cladding failure to the plenum gas inventory and to temperature increases during transient heating.

o Gas Release and Plenum Pressure Model

Gas retention in metallic fuels tends to saturate after some small burnup bu$_1$. This allows a simple model to be developed to estimate the plenum pressure as a function of burnup. The model is based on the following assumptions:

- The hot dimensions at zero burnup are the same as the fabricated dimensions.
- The plenum is initially filled to one atmosphere at room temperature.
- The initial sodium level is 1 inch above the top of the fuel.
- Cladding swelling and cladding creep are negligible.

- Burnup, swelling and gas release are averaged over the length of the fuel using the peak/average burnup.

- The gas volume Vg available inside the pin equals the total volume inside the cladding tube minus the volume occupied by a) the fuel alloy, b) bond sodium, c) non-gaseous fission products and d) gaseous fission products in closed porosity. The amount of sodium in open porosity or the amount of non-gaseous fission products in the sodium do not matter in this method of bookkeeping as long as assumptions 7 and 8 are valid.

- The fuel is in a state of uniform hydrostatic stress equal to the plenum pressure P.

- Swelling of non-gaseous fission products is calculated using their yield along with the assumption that the atoms occupy their atomic volume within the pin. Swelling is referenced to the fabricated fuel lvolume so that various contributions can be added.

- Gaseous fission products retained in the fuel are in bubbles of radius r_b (treated as a parameter). The gas pressure inside the bubbles is calculated using the reduced van der Waals' equation with a volume per xenon or krypton atom of 85 x $10^{-24}cm^3$. The fission gas yield for U-xPu-10 w/o Zr is approximately 0.9 x 10^{20} atoms per a/o burnup per fabricated cubic centimeter.

- Fuel swelling is isotopic prior to some critical burnup bu_1 (a parameter) for breakaway swelling.

- At bu_1 there are sufficient fission gas bubbles in the fuel so that short-range interconnection occurs. The resultant "breakaway swelling" of the fuel, which is assumed to translate to purely radial deformation closing the fuel-cladding gap, rapidly forms enough porosity to give long-range interconnection to the plenum (the theoretical limit for percolation of identical spherical pores is 16% porosity or 20% swelling). The amount of fission gas retained in the fuel saturates to that amount which is present at bu_1. The fractional release to the plenum for $bu \geq bu_1$ is therefore just $1-bu_1/bu$.

- Most of the gas in the free volume V_g is at the temperature of the plenum which is assumed to follow the coolant outlet temperature.

The critical burnup bu_1 was chosen here to give either 100% release (bu_1=0.0) or 90% release (bu_1=1.4) at 14 a/o burnup. The calculated plenum pressure is not a strong function of r_b within a reasonable range of 0.01 to 10 microns. The value chosen of 0.1 microns is consistent with observed bubble sizes. Calculations have shown, as expected, that a 10% change in gas release gives about a 10% change in plenum pressure.

o Sensitivity of Transient Cladding Failure to Plenum Pressure and Temperature

The sensitivity of cladding failure to pressure and temperature can easily be estimated. At end-of-life transient failure of HT-9 cladding is expected to occur by creep-rupture near the top of the fuel where the primary loading is due to the plenum pressure. We therefore consider simple power-law behavior where the equivalent steady-state creep rate $\dot{\epsilon}_s$ is given by the Dorn equation.

$$\dot{\epsilon}_s = C_o \left(\frac{\sigma}{E}\right)^n \exp(-Q_c/RT) \tag{1}$$

and the rupture time t_r is given by

$$t_r = B_o \left(\frac{\sigma}{E}\right)^m \exp(Q_r/RT) \tag{2}$$

In these equations σ is the equivalent stress (equal to $\sqrt{3}/2$ times the hoop stress for biaxial pressure loading), E is the elastic modulus, R = 1.987 is the gas constant, T is the absolute temperature and C_o, B_o, n, m, Q_c and Q_r are (approximately) constants. Furthermore, the Monkman-Grant relationship

$$\dot{\epsilon}_s t_r \cong \text{constant} = A_o \tag{3}$$

is frequently observed for materials where the rupture process is governed by the same mechanism as that governing deformation. Examples are creep growth of cavities and plastic instability. The Monkman-Grant relationship implies that

$$n = m, \tag{4a}$$

$$Q_c = Q_r, \tag{4b}$$

and

$$B_o C_o = A_o. \tag{4c}$$

The theoretical basis for Eq. (1) is diffusion controlled climb of dislocations past obstacles with an activation energy Q_c equal to the activation energy Q_v for bulk diffusion. This deformation mechanism generally requires that the temperatures T exceed half the absolute melting temperature T_m and also requires σ/E to be less than about 10^{-3}. For HT-9 cladding and the conditions of interest here.

$$T \geq 900K \geq 0.5Tm = 850K,$$

$$E = 2.12 \times 10^5 [1.144 - 4.856 \times 10^{-4} T], \text{MPa}, \tag{5}$$

$$\sigma = \frac{\sqrt{3}}{2}\frac{Pr}{h} \tag{6}$$

where P is the plenum pressure, r is the cladding inner radius and h is the cladding wall thickness. The radius-to-thickness ration for SAFR cladding designs is 5.5 so that at a maximum pressure of about P = 10 MPA and T = 1000K, $\sigma/E \cong 10^{-4}$. Both the stress and the temperatures are therefore within the range of validity of Eq. (1), as long as the primary cladding loading mechanism is the plenum pressure.

Regardless of the data of the form of the creep-rate and stress-rupture correlations, the stress and temperature sensitivities of $\dot{\epsilon}_s$ and t_r are often defined in terms of the following effective exponents and activation energies

$$n_{eff} = \frac{\partial \ln \dot{\epsilon}_s}{\partial \ln \sigma} \Big|_T, \tag{7a}$$

$$\frac{Q_{ceff}}{R} = \frac{\partial \ln \dot{\epsilon}_s}{\partial (1/T)} \Big|_\sigma, \tag{7b}$$

$$m_{eff} = - \frac{\partial \ln t_r}{\partial \ln \sigma} \Big|_T, \text{ and} \tag{7c}$$

$$\frac{Q_{reff}}{R} = \frac{\partial \ln t_r}{\partial (1/T)} \Big|_\sigma. \tag{7d}$$

For the power-low models given by Eqs. (1) and (2)

$$n_{eff} = n, \tag{8a}$$

$$\frac{Q_{ceff}}{R} = \frac{Q_c}{R} - \frac{nT^2}{E}\frac{dE}{dT}, \tag{8b}$$

$$m_{eff} = m, \text{ and} \tag{8c}$$

$$\frac{Q_{reff}}{R} = \frac{Q_r}{R} - \frac{mT^2}{E}\frac{dE}{dT}. \tag{8d}$$

The sensitivities of rupture time and creep rate to stress and temperature are given by the definitions in Eqs. (7). If the Monkman-Grant relationship is valid, the two sets of parameters are equal. Of interest here is the rupture time. The differential increase in rupture life is given by

$$d(\ln t_r) = \frac{\partial \ln tr}{\partial o} do + \frac{\partial \ln tr}{\partial T} dT \tag{9}$$

or, in terms of the more convenient base 10 logarithms and the definitions in Eq. (7),

$$-d(\log t_r) = 0.434 \left[m_{eff} \frac{d\sigma}{\sigma} + \frac{Q_{reff}}{RT}\frac{dT}{T}\right]. \tag{10}$$

Two methods have been used to calculate transient failure of ternary fuel pins. One is a life fraction method based on a HEDL (Hanford Engineering and Development Laboratory) correlation of data from their transient burst tests on HT-9 cladding. The other is based on a deformation equation developed by ANL (Argonne National Laboratory) from tensile data and validated against the cladding burst test results. The HEDL correlation has an effective activation energy Q_{reff} of 70,000 cal/mole and a power m_{eff} of 4.7 while the ANL correlation has an effective activation energy of 76,000 cal/mole and a power of 2.26.

Calculations of transient rupture time at end-of-life show that the rupture life assuming 100% fission gas release to the plenum is approximately 0.8 of the rupture life assuming 90% release. This decrease in rupture life is consistent with deformation-based failure results and is well within the normal scatter in rupture data. The life fraction method gives a similar small sensitivity of rupture life to a 10% increase in plenum pressure. This sensitivity is much less than the temperature sensitivity which, from Eq. 10, is about an order of magnitude change in transient rupture time for a 100°F (55K) change in temperature.

4.ANL 4

Please expand the test data provided in the viewgraph on fuel and blanket irradiation data and programs presented in the meeting to include, where available, linear heating rate, clad thickness and clad temperature. This information should be provided for those tests already completed and for those tests currently planned and should cover steady and transient test programs.

Response

See attached updated ANL tables of:

- o Status of current metallic fuel irradiations
- o Planned metallic fuel irradiation program in EBR-II
- o Metallic fuel irradiation program in FFTF

METALLIC FUEL IRRADIATION STATUS

EXPERIMENT	Pu CONTENT	CLADDING	PIN O.D.	PIN Wall	Kw/ft*	Peak Clad* Temp °F(°C)	CURRENT BURNUP,at.%	STATUS
EBR-II X419B	0, 8, 19	D9	0.230	0.015	13.8	1040(560)	8.9	PIE at 1,2,3 at.%; RTCB
EBR-II X420A	0, 8, 19	D9	0.230	0.015	14.7	1076(580)	9.3	PIE at 6 at.%; RTCB
EBR-II X421	0, 8, 19	D9	0.230	0.015	13.8	1040(560)	10.0	PIE at 10 at.%; RTCB
EBR-II X423C	0,3,8,19,22,26	316	0.290	0.016	12.6	972(522)	3.5	PIE at 0.5,0.9,2,(5) at.%
EBR-II X425A	0, 8, 19	HT9	0.230	0.015	14.7	1112(600)	5.2	PIE at 3,(6) at.%; RTCB
EBR-II X428	8, 19	316	0.174	0.012	7.0	1080(582)	2.5	Complete
EBR-II X429	0, 8, 19	HT9/316	0.230	0.015	14.7	1112(600)	2.4	PIE at (8) at.%; RTCB
EBR-II XY24	19	316	0.174	0.012	6.9	1036(558)	2.5+ 40 days	Goal 2.5+ ~140 days
EBR-II XY27	8	316	0.174	0.012	6.9	970(521)	2.5+ 0 days	Goal 2.5+ ~256 days
EBR-II X430	0, 19	HT9	0.290	0.016	15.0	1157(625)	0	Starts in May
FFTF IFR-1	0, 8, 19	D9	0.270	0.022	15.0	1140(615)	2.8	Goal of 10 at.%
FFTF MFF1&1A	0	HT9	0.270	0.022	13.5	1127(608)	0	Starts in June

* BOL Design Values

PLANNED METALLIC FUEL IRRADIATION PROGRAM IN EBR-II

EXPERIMENT	CLADDING	PIN O.D.	DESCRIPTION	CYCLE	DATE
Mark-III: X435 X436, X437, X438	D9	0.230	Four assemblies of U-10Zr RTCB to qualify driver fuel	(1) 144 (3) 145	5/87 9/87
Mark-IIC/CS: XY-26	316	0.174	Qualification of U-10Zr fuel for control and safety rods	145	9/87
Design Parameters: X441	HT9/D9	0.230	Fuel design variables of smear density, plenum to fuel, thickness, and wt% Zr	145	9/87
High Plutonium	HT9/D9	0.290	Characterization of fuel containing 25-30 wt% plutonium	146	11/87
Mark-III RBCB	D9	0.230	Series of midlife extended RBCB on U-xPu-10Zr fuel for x=0, 8, and 19	146	11/87
Fuel Operating Variables	HT9	0.230/0.270	Operating conditions: 10-25 Kw/ft, and 550 to 650°C	146	11/87
Blanket Design Variables: X431-X432	HT9	0.370	Smear density: 85-90% for U-xZr where x=2, 6, 10	147	2/88
Peak Normal Temperature	HT9	?	Performance evaluation of HT9-clad U-10Zr fuel at 650°C peak cladding midwall temperature	147	2/88
Fuel Fabrication Variables II	HT9	0.230	Relax fuel specification with minimal sodium bond	147	2/88
Mark-IV	HT9	0.230	Four assemblies each of U-10Zr (and U-19Pu-10Zr RTCB to qualify driver fuel?)	148	5/88
Mark-IV Offnormal	HT9	0.230	Standard assembly with multiple recons to generate offnormal test elements	148	5/88
Blanket Operating Temperatures	HT9	0.370	Simulated increasing temperature with increased orificing and power (by position)	148	5/88
Mark-III Overtemperature: XY-25	D9	0.230	RTCB at elevated temperature to determine lifetime above fuel-cladding eutectic formation temperature	?	11/88

PLANNED METALLIC FUEL IRRADIATION PROGRAM IN EBR-II

EXPERIMENT	CLADDING	PIN O.D.	DESCRIPTION	CYCLE	DATE
Blanket Overtemperature	HT9	0.370	RTCB at elevated temperature to determine lifetime above fuel-cladding eutectic formation temperature	?	5/89
Mark-IV Simulated Recycle	HT9	0.230	Performance testing of simulated reprocessed fuel	?	6/89
Mark-IV Overtemperature	HT9	0.230	RTCB at elevated temperature to determine lifetime above fuel-cladding eutectic formation temperature	?	11/89
Mark-III/Mark-IV Temperature-to-melt	HT9	0.230	Test to determine melt fraction as a function of composition and burnup	?	11/89
Mark-III/IV Overpower	D9/HT9	0.230	Operational Overpower transient	?	11/89
Instrumented Fuel	HT9	0.290	Characterization of Thermal response and fuel performance	?	5/90
Blanket Overpower	HT9	0.370	Operational overpower transient	?	11/90
Fuel FCMI	HT9	---	Characterization of FCMI	?	?
Fuel Minor Impurities	HT9	---	Characterization of fuel performance to relax fuel specification	?	?

METALLIC FUEL IRRADIATION PROGRAM IN FFTF

EXPERIMENT	CLADDING	PIN O.D.	POWER Kw/ft	PEAK CLAD TEMP. °C	DESCRIPTION	FFTF CYCLE	START DATE
IFR-1	D9	0.270	15	615	Lead ternary fuel	9A	9/86
MFF-1	HT9	0.270	13.5	608	Lead HT9/U-10Zr	9C	6/87
-1A	HT9	0.270	13.5	608	Lead HT9/U-10Zr	9C	6/87
MFF-2	HT9	0.270	17	600	Series III Qual: Normal Temp.	10B	2/88
-3	HT9	0.270	17	650	Series III Qual: High Temp.	10C	7/88
4-5	HT9	0.270	17	600	Series III Qual: Normal Temp	11A	12/88
-6	HT9	0.270	17	600	Series III Qual: Normal Temp	11B	5/89
7-10	HT9	0.270	17	600	Series III Qual: Normal Temp	11C	10/89
11-14	HT9	0.270	17	600	Series III Qual: Normal Temp	12C	1/91
Lead Series III Drivers:	HT9	0.270	15	600	Standard		
1						10C	7/88
3						11B	5/89
4						11C	7/89
4						12A	3/90
4						12B	8/90
4						12C	1/91
16						13	6/91
10						14	9/92

4.ANL 5

Provide a description of the models and correlations used in the "LIFE" code to predict metal fuel burnup capability. Include a discussion of uncertainties in the models and correlations and how these are incorporated in your predictions of burnup capability for both fuel and blanket.

Response

o Description of Models and Correlations Used in LIFE-METAL to Predict Metal Fuel Burnup Capability

Progress on LIFE-METAL development, verification, validation, and documentation is reported in the "IFR Fuels Performance and Fabrication" progress reports, in ANL-IFR reports, and in open literature publications. Table 1 summarizes the titles of the ANL-IFR reports, and the open literature publications. A status report describing models, correlations, properties, verification, validation (up to ~10 at.% burnup), input description, and output description is being prepared for publication in September 1987.

Until publication of the LIFE-METAL status report, a detailed description of models and correlations is not available in any convenient, easily-readable form. Reference 2 has some description of the fuel fission gas release and swelling models. All work will be reported in a systematic fashion in the September 1987 status report.

o Model/Correlation Uncertainties

Uncertainties in properties correlations are documented in ANL-IFR-29 (Ref. 6). The major uncertainties in LIFE-METAL are associated with the models, the model parameters, and the interaction of the models. The procedure adopted in this area is to compare code predictions to integral, in-reactor fuel element data. For example, parameters routinely determined from fuel element irradiations are fuel length change, fuel volume change, fission gas release, and cladding strain. Other parameters measured on a more selective basis are: local porosity distribution, local alloy constituent weight fractions, and fuel/cladding metallurgical interaction. With this approach, an estimate of overall code uncertainty in performance predictions is obtained, rather than a set of uncertainties for individual models.

With regard to thermal performance, code calculations for fuel temperatures are compared to calculated solidus temperatures to determine a design margin. The uncertainties in the fuel temperature calculation arise from input parameters (e.g., coolant outlet temperature, linear power, and local cladding hot spots), calculated parameters which are used in the fuel thermal conductivity correlation (e.g., alloy constituent concentrations and local fission-gas porosity), the accuracy of the correlation itself, and fuel behavior not explicitly modeled (e.g., partial sodium logging and fuel cracking). Most of these uncertainties are incorporated into conservative (2σ) input operating conditions. In turn, an uncertainty is incorporated into the fuel solidus temperature. This uncertainty ranges from a low of − 40°C for the U-Pu system with small concentrations of Zr to a high of ±125°C for mid-composition ranges of U-Pu-Zr. The approach of comparing the upper bound (e.g., 2σ) fuel temperatures to the lower bound solidus temperature tends to be overly conservative. If this approach leads to an unacceptable design margin, then a better statistical treatment is warranted. Also, it suggests the need for better U-Pu-Ar solidus data to reduce the $\pm$ 125°C uncertainty. For more details on this subject, the reader is referred to the ANL memo "Power-to-Melt Calculations for U-19Pu-10Zr Fuel Rods from EBR-II Experimental Subassemblies X419, X420 and X421."

In terms of predicting burnup capability for metal fuel and blanket rods, the major uncertainties are in predicting cladding temperatures, cladding degradation due to fuel/cladding metallurgical interaction, plenum pressure, and fuel/cladding mechanical interaction. Uncertainties in cladding temperatures are accounted for in the standard way of using 2σ operating conditions as input. Cladding degradation due to fuel/cladding metallurgical interaction is an area of high uncertainty requiring better modeling, more experimental data under controlled surface conditions and gas environment, and standard fabrication and handling techniques. Plenum pressure depends on plenum temperature, moles of fission gas released to the plenum, and plenum volume. Uncertainties in plenum temperature are treated in the same manner as uncertainties in cladding temperatures. Overall gas release is a predicted and measured quantity which can be assigned an uncertainty factor. Effective plenum volume is a more subtle issue. The calculated plenum volume depends on interior cladding-tube volume, fuel volume, sodium volume, and the degree to which interconnected fuel porosity volume acts as part of the plenum volume. Currently, it is conservatively assumed that no sodium logging occurs and that none of the interior fuel volume accommodates plenum gases.

With regard to fuel/cladding mechanical interaction (FCMI) there is no evidence of well-designed (e.g., 75% smear density) U-5Fs or U-Pu-Zr fuel causing cladding failure. Some evidence exists which <u>suggests</u> that FCMI may contribute to cladding strain in the lower 25% of the fuel column. However, uncertainties in cladding swelling and creep strain correlations, as well as fuel properties, have made it difficult to quantify FCMI. The DP1,2 experiment (see memo "LIFE-METAL Calculations in Support of Proposed Irradiation Tests DP-1 and DP-2") is designed to include initial smear densities of 70-85%. Data from this experiment will help to validate the LIFE-METAL mechanical models/correlations for fuel and cladding.

The final set of uncertainties are associated with cladding design and failure criteria. This subject is documented in the ANL memo "Peak Design and Lifetime Burnups vs. Peak Cladding-Midwall Temperatures for Planned (MFF-3) U-10Zr/HT9 Elements in FFTF."

o Summary of LIFE-METAL Status

Because of the evolving nature of LIFE-METAL, no single report is complete or up-to-date. The reports are listed in Table 1 and are available upon request. The planned date for a comprehensive report on LIFE-METAL is September 1987. The flavor of LIFE-METAL development and validation is contained in Reference 2. Because of page limitations the amount of detail presented in this paper is limited. Also, only a very limited data base was available for validation at the time this paper was written. A more extensive validation to data up to 5.5 at.% burnup is contained in Tables 3 and 4 of Reference 1. The planned September 1987 report will include validation up to 10 at.% burnup.

Table 1. Summary of LIFE-METAL Documents

Open Literature Publications

1. M. C. Billone, "LIFE-METAL Analysis of U-Pu-Zr Fuel Performance," 89th Annual Meeting Abstracts, A. Cer. S. April 26-30, 1987, 15-N-87, p. 283.

2. M. C. Billone, Y. Y. Liu, E. E. Gruber, T. H. Hughes, and J. M. Kramer, "Status of Fuel Element Modeling Codes for Metallic Fuels," proceedings of the ANS International Conference on Reliable Fuels for Liquid Metal Reactors, Tucson, AZ, September 7-11, 1986, pp. 5-77 to 5-92.

ANL-IFR Reports

3. H. C. Tsai and M. C. Billone, "Input to the Test Design Description (TDD), Volume 1A, Design Description and Safety Analysis for MFF-1 Metal Fuels Irradiation Test in FFTF," to be published as an ANL-IFR report (February 1987)

4. Y. Y. Liu and M. C. Billone, "Thermoelastic Stresses in U-Pu-Zr Ternary Alloy Fuels-an Analytical Interpretation of Results in the IFR Lead Irradiation Experiments," Argonne National Laboratory, ANL-IFR-49, August 1986.

5. H. Tsai, L. A. Neimark, M. C.Billone, R. M. Fryer, J. F. Koenig, W. K. Lehto, and D. J. Malloy, "Test Design Description (TDD), Volume 1A, Design Description and Safety Analysis for IFR-1 Metal Fuels Irradiation Test in FFTF," Argonne National Laboratory, ANL-IFR-33, January 1986.

6. G. L. Hofman, L. Leibowitz, J. M. Kramer, M. C. Billone and J. F. Koenig, "Metallic Fuels Handbook," Argonne National Laboratory, ANL-IFR-29, November 1985.

4.ANL 6

Data from fully prototypic fuel (material composition, geometry, irradiation) will not be available in the near term. Describe the data base as it will be developed over the next few years and how it will be used to support the fuel design, safety evaluation and licensing efforts. This should include both operational and transient response characteristics.

Response

The data base for metallic fuels will expand greatly in the next two years. The highlights will feature the first end-of-life breach which will identify lifetime potential and breach mechanism; the conversion of EBR-II to a U-Zr/U-Pu-Zr core and FFTF to a U-Zr core; the expansion of off-normal testing to include, at moderate burnups, RBCB, overtemperature and overpower events; and a broad base of properties data characterizing physical, thermal-dynamic, and materials behavior. The tables provided in the response to comment 4.ANL 4 identify the tests planned in EBR-II and FFTF with the proposed start dates. These tests will complement those currently in progress.

The current plan includes testing of high plutonium bearing alloys, testing at a peak cladding temperature of 650°C in EBR-II and FFTF, and more off-normal testing at an earlier date than previously planned. Key tests have been planned to specifically provide data for the calibration and verification of LIFE-METAL. The first of these is currently being fabricated.

Special off-normal test periods in EBR-II have been defined for November of each year. In 1988 and 1989 overtemprature, overpower, and temperature-to-melt tests are planned.

These in-reactor tests will be supported by post-irradiation performance evaluations of the lead tests and associated laboratory tests. In particular, overtemperature tests of irradiated fuel sections will establish the safety margins for conducting the in-reactor tests and define the appropriate test matrix. Likewise, fuel compressibility tests will quantify the significance of additional fuel cladding mechanical interaction which may be operative during transient overpower events.

The irradiation program is sharply focused to address the important concerns. The designs, material and test conditions envelop completely the current innovative designs. Both EBR-II and FFTF testing will provide the needed data base to support design, safety evaluations, and licensing.

GE is concerned that the irradiation test plan, as proposed by ANL, does not contain sufficient testing of the prototypic PRISM ternary fuel (U-126.4Pu-10Zr) to provide full validation of this specific fuel form. GE has recommended to ANL and DOE that the irradiation program be altered to include more PRISM prototypic fuel.

__4.ANL 7__

Provide a discussion of the models and correlations developed at HEDL to characterize the steady state and transient response of HT-9 cladding. Include the experimental data used to support the models and development of the correlations. The discussion should include the phenomena of swelling, irradiation and thermal creep, yield strength and stress rupture. Provide the information as functions of cladding thickness and diameter where such data is available.

__Response__

Correlations for describing physical and mechanical properties of alloy HT-9 are discussed in a Westinghouse Hanford Company report entitled "Physical and Mechanical Properties of Alloy HT-9 Used in the Design Analysis of the Core Demonstration Experiment," (HEDL-TC-2845). The properties described in this report are: density, thermal expansion, specific heat, thermal conductivity, modulus of elasticity, shear modulus, Poisson's ratio, yield strength, ultimate strength, uniform elongation, total elongation, swelling, creep and biaxial stress rupture.

The experimental data used to develop the correlations exist in a variety of quarterly reports, topical reports and letters (both internal and external correspondence). A bibliography of these reports and letters is attached and HEDL will provide specific documents on request.

The effects of cladding thickness and diameter were not specifically addressed in the experiments conducted on HT-9. A few limited stress rupture tests were conducted on different sizes of cladding, but these data were not used for the correlation in HEDL-TC-2845.

HT9 DATA BASE

Quarterly Reports

Quarterly reports on the cladding and duct materials work were initiated in 1974. The title of the report was "Alloy Development Program Quarterly Technical Progress Report". The first report number was TC-160-1. In 1977, the title was changed to "National Cladding/Duct Materials Development Program Quarterly Technical Progress Letter"; the number system remained unchanged. In 1983, the report numbering system was altered to read HEDL-TC-160-XX. The last quarterly report, HEDL-TC-160-47, was issued in 1985. In the listings below, only the last number of the report is given, i.e., the "XX" portion of the HEDL-TC-160-XX number.

I. HT9 - CREEP

MOTA Data - FFTF

46 R. J. Puigh, "Interim Report on the MOTA Creep Data from MOTA-1B and MOTA," p. 1-108.

40 R. J. Puigh, "In-Reactor Creep and Creep Rupture Data from MOTA-1A Irradiated in FFTF," p. 19-108.

Creep in Bending Data

40 J. M. Rosa, T. Lauritzen and S. Vaidyanathan, "Advanced Alloy Creep in Bending at High Fluence," p. 3-18.

37 J. M. Rosa, "Advanced Alloy Creep in Bending at 12 x 10^{22} n/cm^2," p. 41-52.

30 J. M. Rosa, "Advanced Alloy Creep in Bending at 6 x 10^{22} n/cm^2," p. 103-109.

23 A. J. McSherry and M. Patel, "Analysis of Results from the Second Examination of the Advanced Alloy Creep in Bending Test," p. 11-16.

21 A. J. McSherry and M. Patel, "Results of the First Interim Examination of the Advanced Alloy Creep-in-Bending Test," p. 3-7.

EBR-II Data

36 R. J. Puigh, "Interim Report on the Creep Results from the AAXIV Experiment," p. 3-100.

35 R. J. Puigh, "A Comparison of the 316 SS, D9C1, and HT9 In-Reactor Creep Data from the AAXIV - Part 1 Experiment," p. 79-88.

32 R. J. Puigh, "An Empirical Correlation for the Low Temperature In-Reactor Creep Behavior of HT9," p. 33-49.

30 R. J. Puigh, "Creep Behaviors of Selected Ferritic Alloys," p. 17-26.

30 R. J. Puigh, "High Fluence In-Reactor Creep of Advanced Alloys," p. 27-59.

28 R. J. Puigh, "Candidate Advanced Alloy In-Reactor Creep," p. 11-30.

26 E. R. Gilbert and B. A. Adams, "Candidate Advanced Alloy In-Reactor Creep," p. 21-31.

I. HT9 - CREEP (Cont'd)

EBR-II Data (Cont'd)

16 B. A. Chin and E. R. Gilbert, "In-Reactor Creep of Advanced Alloys," p. 3-63.

16 B. A. Chin and E. R. Gilbert, "The Fluence and Temperature Dependence of Creep in Commercial Advanced Reactors," p. 78-105.

Thermal Creep Data

43 R. J. Puigh, "Thermal Creep Equations for HT9," p. 22-42.

18 B. A. Chin and E. R. Gilbert, "Thermal Creep of Commercial Advanced Alloys," p. 33-46.

II. HT9 - SWELLING

37 D. S. Gelles, R. J. Puigh, J. Pintler and R. L. Meinecke, "Density Change Measurements on AISI 316 and Selected Ferritic Alloy Specimens," p. 11-24.

36 R. J. Puigh, "Interim Report on the Creep Results from the AAXIV Experiment," p. 3-100.

33 D. S. Gelles, "Density Change Measurements on Irradiated Simple Ferritic Alloys," p. 87-90.

32 D. S. Gelles, L. Thomas, D. Peterson, "Microstructural Examination and Density Determinations for Irradiated Ferritic and Martensitic Alloys," p. 105-123.

27 F. A. Smidt and J. R. Reed, "Microstructural Observations of Ferritic Alloys at AAI Interim Examination at 1.6 x 10^{23} n/cm^2," p. 71-88.

22 J. Bates, "Development of Candidate Alloy Swelling and Thermal Densification," p. 37-46.

16 J. Bates and R. R. Borisch, "Swelling in the Ferritic Alloys HT9 and D57," p. 200-205.

III. TENSILE PROPERTIES

46 T. Lauritzen, et al., "Tensile Properties of Alloys HT9 and Modified 9Cr-1Mo Irradiated in AAXV," p. 147-159.

45 T. Lauritzen, et al., "Tensile Properties of Irradiated HT9 Weldments," p. 51-63.

44 T. Lauritzen, et al., "Some Effects of Irradiation on the Tensile Properties of Prototypic HT9 Weldments," p. 62-66.

42 T. Lauritzen, et al., "Tensile Properties of Reactor-Irradiated HT9 Weldments," p. 49-53.

41 M. L. Hamilton and C. Martinez, "Tensile Properties of Irradiated HT9," p. 19-27.

39 T. Lauritzen, et al., "Mechanical Properties of Reactor-Irradiated HT9: Effect of Thermomechanical Treatments," p. 13-25.

35 N. F. Panayotou and M. L. Hamilton, "Tensile Test Results for D9-C1 and HT9 Irradiated in AA-XIV," p. 161-170.

34 J. R. Hawthorne, "Tensile Property Determinations for Irradiated Ferritic Alloys from Experiment AA-XIV," p. 113-126.

34 T. Lauritzen, et al., "The Tensile Ductility of Reactor Irradiated HT9," p. 145-152.

33 T. Lauritzen, et al., "Tensile Properties of Reactor-Irradiated HT9," p. 127-130.

33 J. R. Hawthorne, "Postirradiation Tensile Property Determinations for Ferritic Alloys from Experiment AA-XIV," p. 131-140.

27 M. L. Hamilton and B. Mastel, "Mechanical Behavior of Advanced Alloys," p. 321-328.

24 J. A. Horak, et al., "Mechanical Properties of Advanced Alloys Irradiated to Fluences up to 10^{23} n/cm^2 (E>0.1 MeV)," p. 275-296.

16 A. F. Rowcliffe, et al., "Postirradiation Tensile Testing of Candidate Alloys," p. 383-411.

IV. FRACTURE BEHAVIOR (COMPACT TENSION AND CHARPY)

46 F. H. Huang, "Fracture Properties of HT9 Irradiated to 9 x 10^{22} n/cm^2," p. 160-170.

44 W. L. Hu, "Charpy Impact Test Results of Ferritic Stainless Steel Alloys Irradiated in the AA-XV Phase II Experiment," p. 24-33.

41 F. H. Huang, "The J_{Ic} Fracture Toughness Transition Behavior of HT9," p. 28-38.

37 F. H. Huang, "Fracture Toughness of Ferritic Alloys from the AA-XIV Phase II Experiment," p. 65-72.

36 W. L. Hu, "Charpy Test Results on HT9 Alloys Irradiated in the AA-XIV Phase II Experiment," p. 217-230.

35 F. H. Huang, "Fracture Toughness of HT9 Irradiated to a Fluence of 6 x 10^{22} n/cm^2," p. 147-152.

34 F. H. Huang, "Post-Irradiation Fracture Response of Ferritic Alloys to Various Thermomechanical Heat Treatments and Compositions," p. 127-144.

33 F. H. Huang, "Post-Irradiation Fracture Toughness of Alloy HT9," p. 105-114.

29 F. H. Huang, "Fracture Resistance of Irradiated HT9 and 9Cr-1Mo," p. 367-378.

28 J. R. Hawthorne, "Postirradiation Fracture Resistance Determinations for Ferritic Alloys from the AA-XV Experiment," p. 171-178.

28 F. H. Huang, "Fracture Toughness of Irradiated HT9," p. 179-188.

27 J. R. Hawthorne, "Fracture Resistance Testing of Alloy HT9 and Other Ferritic Stainless Steels in the Unirradiated (Reference) Condition," p. 259-266.

21 F. A. Smidt and J. R. Hawthorne, "Evaluation of Fracture Toughness and Tensile Properties of Irradiated HT9," p. 253-261.

20 J. R. Hawthorne, et al., "Fracture Behavior of Ferritic Alloy HT9 After Irradiation," p. 235-240.

19 J. R. Hawthorne, et al., "Fracture Testing of Ferritic Alloy HT9 After 5000 Hour Thermal Aging at 427 and 538°C," p. 159-168.

V. STRESS RUPTURE

In-Reactor

45 R. J. Puigh, "In-Reactor Stress Rupture Data from MOTA After Peak Fluence of 17 x 10^{22} n/cm^2," p. 26-39.

44 R. J. Puigh, "In-Reactor Stress Rupture Data from MOTA After Peak Fluence of 13 x 10^{22} n/cm^2 (Through Cycle 5 Irradiation)," p. 1-23.

42 R. J. Puigh, "In-Reactor Stress Rupture Data from MOTA After Peak Fluence of 9.6 x 10^{22} n/cm^2 (Through Cycle 4 Irradiation)," p. 2-20.

40 R. J. Puigh, "The In-Reactor Creep and Creep Rupture Data from MOTA 1A Irradiation in FFTF," p. 19-108.

Thermal

43 M. L. Hamilton and D. S. Gelles, "Heat Treatment Optimization for Increased Rupture Strength in HT9," p. 1-21.

31 M. L. Hamilton and W. F. Brizes, "Stress Rupture Correlation for Unirradiated HT9," p. 203-208.

27 M. L. Hamilton and D. R. Duncan, "Stress Rupture Behavior of Candidate Alloys," p. 267-274.

VI. TRANSIENT BURST (FCTT)

30 D. R. Duncan and W. F. Brizes, "Transient Behavior of Unirradiated HT9 Tubing," p. 227-236.

VII. SODIUM CORROSION

34 R. P. Anantatmula, "Sodium Compatibility of HT9 and Fe9Cr-1Mo at 600°C," p. 165-183.

30 R. P. Anantatmula, "Sodium Corrosion Characterization of HT9 at 650°C," p. 267-282.

WHC TOPICAL REPORTS ON HT9

1. M. L. Hamilton, et al., "HT9 Cladding Characterization," TC-1898, January 1981.

2. D. S. Gelles and J. L. Straalsund, "Assessment of the Performance Potential of HT9 for LMFBR Ducts," TC-1996, June 1981.

3. R. P. Anantatmula, "Sodium Corrosion Characterization of HT9 at 650°C," TC-2038, August 1981.

4. N. S. Cannon, "Correlations for HT9 Cladding Flow, Failure and Thermal Expansion for Transient Analysis," HEDL-TC-2557, April 1984.

5. M. L. Hamilton, "Thermal Expansion of HT9," HEDL-TC-2667, February 1985.

6. M. L. Hamilton and N. S. Cannon, "HT9 Transient Data Base and Failure Correlation," HEDL-TC-2681, March 1985.

7. F. H. Huang, et al., "The Static and Impact Fracture Behavior of Irradiated HT9," HEDL-TC-2456, September 1983.

8. D. R. Precechtel, et al., "Development of HT9 Alloy Duct Attachment," HEDL-TC-2293, February 1984.

9. J. L. Ethridge, "Evaluation of a Mechanical Attachment for HT9 Ducts," HEDL-TC-2299, January 1983.

10. A. E. Bridges, "Evaluation of a Mechanical Attachment for HT9 Ducts," HEDL-TC-2299, Addendum 1, April 1984.

GE TOPICAL REPORTS ON HT9

1. T. Lauritzen, et al., "The Pre-Irradiation Characterization of the AAXIV Ferritic Alloys," GEFR-00575, September 1981.

WHC CORRESPONDENCE ON HT9

Letters with External Distribution

1. "Stress-Free Swelling Equations for HT9 and D68," Corres. No. 7952616, July 25, 1979.
2. "Latest Edition of the NSMH Pages for the In-Reactor Creep Behavior of HT9," Corres. No. 8153257, September 21, 1981.
3. "Creep Behavior of Alloy HT9," Corres. No. 8154089, November 30, 1981.
4. "Tensile Correlations for HT9," Corres. No. 8250159, January 15, 1982.
5. "Stress Rupture Correlations for D9 and HT9," Corres. No. 8250399, February 1, 1982.
6. "HT9 Mechanical Properties," Corres. No. 8250772, March 9, 1982.
7. "Fracture Toughness of HT9," Corres. No. 8251794, May 14, 1982.
8. "NSMH Swelling Equation for HT9," Corres. No. 8252411, July 1, 1982.
9. "Proposed In-Reactor Creep Correlation for HT9," Corres. No. 8453204, September 28, 1984.
10. "HT9 FCTT Data," Corres. No. 8551005, April 2, 1985.
11. "DBTT of Irradiated Alloy HT9," Corres. No. AMMLB81-47, May 6, 1981.
12. "Proposed NSMH Tensile Property Equations for HT9," Corres. No. FMTFAWFB206, August 24, 1981.
13. "Proposed HT9 Unirradiated Stress Rupture Equation for the NSMH," Corres. No. FMTFAWFB207, September 17, 1981.
14. "Proposed HT9 Unirradiated Stress Rupture Equation for the NSMH," Corres. No. FMTFAWFB210, January 6, 1982.
15. "NSMH Equations for HT9," Corres. No. XL494-10035, July 29, 1981.

Internal WHC Letters

1. N. S. Cannon to G. D. Johnson, "Tensile Test Results on Irradiated HT9 Cladding," April 24, 1987.
2. G. D. Johnson to J. J. Holmes, "Fracture Toughness of Irradiated HT9," April 21, 1987.

WHC CORRESPONDENCE ON HT9 (Cont'd)

Internal WHC Letters (Cont'd)

3. F. H. Huang to M. L. Hamilton, "Transient Test Results on the AAXV Tube Specimens of HT9 and the High Burnup Cladding from P44," February 4, 1987.

4. F. H. Huang to M. L. Hamilton, "Additional Transient Test Results on HT9 Cladding and a Revised Failure Correlation," December 19, 1986.

5. M. L. Hamilton to G. D. Johnson, "Charpy Impact Behavior of HT9 Irradiated in FFTF to High Fluence" October 2, 1986.

6. M. L. Hamilton to G. D. Johnson, "Mechanical Properties of HT9 in Improved Heat Treatment," October 2, 1986.

7. R. J. Puigh to R. W. Powell, "Report on HT9 Creep and Creep Rupture Data from MOTA 1D," September 30, 1986.

8. M. L. Hamilton to R. D. Leggett, "Recent FCTT-CDF Correlation," August 15, 1986.

9. F. H. Huang to R. D. Leggett, "Transient Tests on HT9 Cladding Irradiated in P44," June 2, 1986.

10. M. L. Hamilton to R. D. Leggett, "Control Milestone I Satisfied for AKL (Summary of Ramp and Hold Results from HT9 Cladding)," February 19, 1986.

11. M. L. Hamilton to W. F. Brown, "Stress Rupture Testing of Pulse Magnetic Welds on HT9 Cladding," March 4, 1985.

12. G. D. Johnson to C. W. Hunter, "FY-84 FCTT Tests," March 4, 1985.

5

RESPONSES TO NRC COMMENTS ON
PRISM PSID CHAPTER 5

5.1 Comment

How did you determine that the UIS will not vibrate during operating conditions?

Response

Flow-induced vibrations are not expected in the UIS based on a survey of potential excitation mechanisms and comparison of the characteristic frequencies of the excitation mechanisms with the UIS component structural frequencies. The two are sufficiently separated to preclude resonance or fluid-elastic feedbacks as shown in the table below. The excitation frequencies in the table are conservatively based on the core assembly fluid velocity of 20 ft/sec and do not take credit for the velocity reduction to about 5 ft/sec in the outlet plenum. These analytical predictions will be verified through testing in subsequent design phases.

UIS FLOW-INDUCED VIBRATION POTENTIAL

COMPONENT	TYPE OF EXCITATION	EXCITATION FREQUENCY, Hz	COMPONENT FREQUENCY, Hz
shroud tube	vortex shedding	11	20
instrument post	vortex shedding	100	300
entire UIS	jet impingement	1.2	3

5.2 Comment

Because containment vessels and guard vessels must be used to ensure the retention of sodium, it can be difficult to determine that a leak has been developing or has occurred. How do you plan to monitor for leaks or for signs of cracking? How will you locate a leak?

Response

The reactor vessel is enclosed by the containment. Leakage of sodium or reactor cover gas will be detected by continuous monitoring. The gap between the reactor vessel and containment vessel will be continuously monitored three ways by 1) sodium liquid detectors, 2) sodium aerosol detectors, and 3) containment vessel pressure. Filters and gas samples will be analyzed to verify a leak indicated by one of the continuous monitoring devices. If a leak is verified the reactor will be shut down and remote viewing with a small TV camera inserted through access ports will be used to identify the size and location of the leak.

5.3 Comment

What plan exists to replace the seal arrangement for the rotatable plug in the event of damage during refueling or wear?

Response

The seals most likely to suffer damage or wear are the dynamic seals used during plug rotation for refueling. The dynamic seals are at the top of the joint between the rotatable plug and the closure (Figure 5.2-3). There are two dynamic seals; they are inflatable elastomer rings and they are buffered.

Below the dynamic seals, in order, are:

a. the primary seals - two buffered, static, inflatable elastomer rings,

b. a maintenance seal space, and

c. a contact seal formed between the plug and its support ledge (during operation) in the closure opening.

In addition, each pair of flanges at the joint has a double O-ring seal between the horizontal faces, and over the entire joint a welded cover is in place during operation.

Replacement of a dynamic seal or a high flange seal requires only the resting of the plug on its support ledge and the inflation of the primary static seals, after which joint hardware can be removed for access to the faulty seal.

Replacement of the static primary seals or the lowest of the flange seals could require the installation of an elastic and inflatable maintenance seal in the space provided. The need for a maintenance seal installation would be dependent on the efficiency of the ledge seal.

The ledge seal has no parts to be replaced. Replacement of the welded cover is a standard operation after refueling.

5.4 Comment

The EM pump

a. How much operating knowledge is available on this size of a pump?

b. What is the life expectance of these pumps?

c. Has ANL made progress on the higher temperature insulation needed for this pump?

d. If the RV is filled with sodium will the pump continue to work?

e. What are the temperature limits?

f. Could contamination decrease the pumps performance?

g. Is the pump's synchronous converter seismically isolated so that it will function during an SSE?

5.4 Comment (Continued)

h. Why not just add a flywheel to the synchronous coverter since the peak fuel temperature in the fuel is directly related to this parameter during a TOP with a scram (as shown in the EBR-II tests)?

i. What are the possible common cause failure modes for the four EM Pumps and what steps have been taken to reduce the likelihood of common cause failure? Are the pumps vulnerable (e.g., electrical insulation) at the high temperature of a LOHS event?

5.4 Response

a. EBR-II pump has a 6500 EM pump in the intermediate sodium group gpm and it has been satisfactorily running for over 23 years.

b. The design life of the PRISM EM pump is thirty years based on very conservative insulation life data and the experience of induction motors in nuclear service.

c. ANL has accelerated life tests of the EM pump insulation in process. Data available projects insulation life in excess of 100 years at reactor operating temperatures.

d. The pump is hermetically sealed and will function so long as it is submerged in sodium.

e. The life expectancy of the pump is limited by the insulation operating temperature. The maximum insulation temperature is 1000°F. Test data indicates that the insulation life may approach 100 years which is well above the pump life goal of 30 years.

f. The primary sodium in the PRISM reactor is expected to be very clean and free of crud. In the event levels become excessive and build up on the pump walls, the sodium velocity for a given capacity level would increase and require additional power because of increased hydraulic losses. The present design has margin available.

g. The synchronous converters are not seismically isolated, however, these machines, used to provide a primary system coastdown on loss of power, are seismically qualified for the SSE event and located below grade in seismic category I vaults (one machine for each of the four EM pumps).

h. The synchronous machine does not presently require a flywheel to provide adequate coastdown. A flywheel will be added if necessary.

i. Simultaneous loss of power to all four pumps is the only common cause failure mode identified. To reduce the likelihood of this common cause failure, physical separation of power supply and power leads to each pump has been accomplished.

5.5 Comment

Could an IHX pipe rupture lead to a completely filled RV and leak sodium past the rotatable plug and into the HAA?

Response

No, an IHX tube rupture would not result in a completely filled reactor vessel. An IHX tube leak would cause a high sodium level in the reactor vessel which in turn would result in reactor scram and trip of the IHTS sodium pump. The pump is tripped to reduce the pressure at the IHX and thereby reduce the leak rate. The IHTS static head is sufficient to assure leakage will be into the reactor vessel. Operator action to mitigate the accident would be; (1) vent the IHTS cover gas to reduce the pressure from 3 to 0 psig and thereby reduce the IHX leak rate, (2) monitor the IHTS sodium level as decay heat is removed by

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5.5 Response (Continued)

natural circulation of the IHTS and dump the IHTS sodium to the drain tank when the level drops below the IHTS pump suction line and natural circulation stops and, (3) after IHTS sodium dump, vent the reactor cover gas to the primary sodium storage tank to maintain the reactor level during the subsequent heatup and cool down to 400°F. For a large IHX leak, the IHTS level could drop fast enough to prevent decay heat removal through the IHTS by natural circulation, as noted above, and the RVACS would continue independently to bring the reactor to the cold shutdown condition. Two IHX leak scenarios are described below; (A) reactor scram with IHTS pump trip and operator corrective action and (B) no automatic or operating corrective action other than a scram on high reactor vessel sodium level. The IHX leak in both cases is assumed to be so large that the decay heat removal is essentially accomplished by RVACS. Case (A) is the expected operational sequence and consequence.

For case (A), the sodium level in the reactor vessel will rise about one foot and the reactor cover gas pressure will increase from 14.7 to 18.7 psia following a major IHX break. The reactor scram will occur at this level. The IHTS pump is shut down, the IHTS cover gas is vented and the IHTS sodium is dumped. About 2700 gallons of IHTS sodium is forced through the IHX break into the reactor vessel. The reactor decay heat is removed by RVACS. During reactor heatup, reactor cover gas is vented to maintain cover gas pressure and level. The level rise is limited to 1.5 feet during the heatup due to thermal expansion of the sodium. About 1000 cubic feet of helium cover gas is vented during the heatup. As the reactor cools to 400°F, cold shutdown temperature, the contracting sodium will reduce the reactor level to the normal range and the vented cover gas will be pulled back into the reactor from the primary sodium storage tank. The net result of this accident is to introduce about 2700 gallons of IHTS sodium into the reactor. The reactor sodium level never drops below the normal level and is never closer than six feet from the reactor top head.

For case (B), the sodium level in the reactor will rise 3.8 feet and the reactor cover gas pressure will increase from 14.7 to 50.2 psia following a major IHX break. The reactor scram will occur after the level raises one foot. When the reactor level increases 3.8 feet, 8900 gallons of sodium will have been pumped from the IHTS into the reactor and the IHTS sodium level will have dropped below the IHTS pump suction. The IHTS pump will then fail due to loss of suction. After pump failure, the reactor cover gas pressure will force 2300 gallons of sodium back through the IHX break into the IHTS. The cover gas pressure will drop from 50.2 to 30.8 psia and come to equilibrium with the static sodium head in the IHTS. The reactor level drops one foot as the sodium is forced into the IHTS. The reactor will then heat up to about 1200°F as RVACS removes the decay heat. During heatup, the reactor level will drop 1.6 feet and the cover gas pressure will increase to 31.1 psia. About 5000 gallons of sodium will be forced through the IHX break into the IHTS as the reactor temperature peaks at about 1200°F. As the reactor is cooled to 400°F, the level increases 1.4 feet and the cover gas pressure decreases to 28.6 psia. About 10,900 gallons of sodium are forced into the reactor from the IHTS during the cooldown. The net results of this accident is to introduce about 12,400 gallons of IHTS sodium into the reactor vessel. The reactor sodium level never drops below the normal level and is never closer than 4.7 feet from the reactor top head.

5.5 Response (Continued)

For both cases, the reactor vessel and closure are designed to accommodate the pressures and temperatures reached.

5.6 Comment

Fuel storage locations are not defined well enough -

a. permanently deformed channels might wobble or just not fit into the storage rack

b. will the nosepiece/receptacle storage rack arrangement allow enough cooling of the assembly?

c. Describe the measures taken to prevent dislodging the spent assemblies.

Response

The fuel storage rack is composed of two parts; the upper storage bracket and the lower storage bracket. The rack is designed to securely hold the fuel element in position in the event of an abnormal occurrence.

The upper bracket supports the vertical weight of the fuel assembly in a socket that matches the interfacing features of the top load pad of the fuel. The bracket is slotted toward the center of the reactor to allow the IVTM to insert and remove the fuel assembly.

The lower bracket prevents any horizontal movement of the lower end of the fuel assemblies with a loose fitting nosepiece receptacle, similar to that used in the core. The loose fit will allow relative horizontal displacement between the upper and lower ends of the assembly.

The storage racks are designed to accept deformed channels by securely supporting the top load pad vertically and horizontally, and loosely supporting the nosepiece horizontally. When a channel is deformed to such an extent that it interferes with another structure, the IVTM can rotate the fuel assembly so that the interference is eliminated.

The design of the storage rack allows the fuel assemblies to be adequately cooled by remaining submersed in the liquid sodium.

5.7 Comment

Operation of the IVTM is not clear in the PSID. Explain in detail how the fuel is moved internal in the core? Note the experiences shared by the Italians and British.

a. Refueling causes the RP to be moved which will wear it down.

b. Steps in the refueling process are unclear.

5.7 Response

The PRISM in-vessel transfer machine (IVTM) is based on the concept used in the Italian PEC and the British PFR fast reactors. The three machines use a single rotating plug and a pantograph mechanism to place the grapple over any core address, as shown in Figure F5.7-1. In addition, PRISM and FFTF have similar refueling machine designs; both use a single rotating plug and an eccentric arm to position the grapple. The main difference between the two designs is FFTF has a fixed length arm and PRISM has a variable length arm via the pantograph linkage.

The PRISM In-Vessel Transfer Machine (IVTM) is used to handle core assemblies in the reactor. The design is a modified pantograph machine with rotary seals. The machine is used only during reactor shutdown and is located in a penetration in the rotatable plug as shown in Figures F5.7-2 and F5.7-3.

The PRISM machine is designed in two parts - above head drive section and the in-vessel section. The junction between the two parts is eight feet above the rotatable plug. The drive section is an electrically driven gear box for operating the in-vessel section. It contains an electric motor, speed reducers, gears, torque limiting clutches, emergency hand operators and other components necessary to provide control and instrumentation. During normal operation of the reactor the drive unit is removed for use on other reactors.

The in-vessel section is positioned vertically from the rotatable plug and extends 39 feet into the reactor. The machine is positioned three feet from the center of the rotatable plug. The machine can be rotated and the pickup leg driven outwards to position the grapple over the required core assembly.

The machine is normally stored, while the reactor is in operation, nearest to the transfer station with the pantograph pickup leg facing toward the transfer station. The machine can be rotated 225 degrees with the stop on the centerline toward the center of the core.

The instrumentation system provides continuous position indication of all machine movement to the control room.

Drive Section (Ex-Vessel)

The drive section (upper part of the IVTM) is a cylindrical unit, approximately 72 inches in height and 15 inches in diameter. An electric motor mounted on the top plate provides, through three gear boxes, all the drives for the complete machine. At the bottom of the unit are seven square drive couplings which mate ith the seven shafts in the top of the lower part. The seven drives consist of five drivers, one orientation control and one sensor. The drives are for:

a. Telescopic tube: raising/lowering - approximately 142 inches

b. Pickup leg: extension/retraction - 36 to 12 inches horizontally

c. Grapple: raising/lowering - approximately 196 inches on the pickup leg

5.7 **Response (Continued)**

d. Machine rotation -- 225 degrees

e. Grapple fingers: 1-1/2 inches - raising to disengage and lowering to engage.

A sixth drive is used to orient the grapple fingers to match the data in the core assembly upper adapters.

A seventh drive is the sensor to indicate the grapple is in full contact with the fuel assembly.

The IVTM rotation is achieved by a pinion engaging a large annular gear mounted in a fixed ring in the outer sleeve. The upper part of the machine sits on the top face of the lower part of the machine, and is located by two drive pins which are of sufficient diameter and length to provide a stable connection between the two parts.

Each of the five shafts have magnetic clutches of the type which energize to engage. When the motor is running, all the gears above the clutches are running. The required drive is selected by energizing the appropriate clutch.

Below the clutches, five of the drives have magnetic brakes. The brakes are energized to release so that in the event of power failure the shafts are held, and they can be released manually for hand-driven operation.

Below the brakes each of the drive shafts is fitted with a torque limiter to provide protection against overloading the mechanisms.

A torque transmitter is mounted in each drive to give continuous indication of the torque being applied.

A hand-operated drive is provided on the drive section for emergency use.

In-Vessel Section

The in-vessel section is positioned vertically through a penetration in the rotating plug and extends 39 feet inside the reactor with the bottom approximately six inches above the core. The lower part consists of four main sections: the main closure and guide assembly, the telescopic tube assembly, pantograph pickup leg and grapple carriage assembly.

The main closure and guide assembly is the top part of the in-vessel section. It is supported in the penetration liner in the rotating plug, and is located by a key fixed to the inside of the penetration. The outer sleeve and inner rotating body are supported and guided by bearings at both ends.

The main guide tube is a thick walled tube slotted to allow the grapple linkage and pantograph to pass through.

5.7 Response (Continued)

The drive shafts are connected through the inner rotating plug to the telescopic tube which runs between sets of diametrically opposite rails in the main guide tube. This system allows the telescopic tube to move up and down.

The telescopic tube is open on one side to coincide with the slot in the main guide into which the pantograph and grapple pickup leg retracts. The pantograph is a parallelogram from which the upper and lower linkage arms are pivoted to the trunnions and the grapple pickup leg. A link connects the center of the linkage arms to the back of the telescopic tube such that the movement of the pickup leg remains horizontal.

The rotation of the pickup leg drive shaft is translated by a lead screw into linear motion to raise or lower the trunnions, thus retracting or extending the vertical pickup leg.

The pickup leg is carried on the outer pivots of the pantograph. It has an internal track to guide the grapple carriage, and bearings at either end to support the lead screw which is used to raise or lower the grapple carriage.

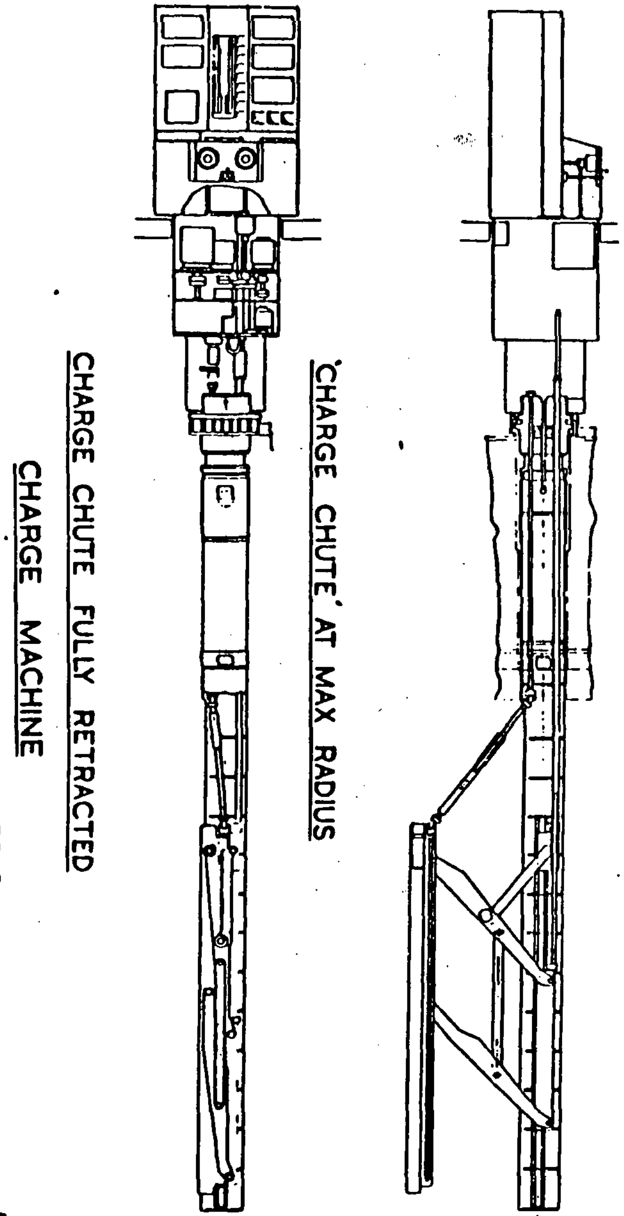
The grapple lead screw is connected to the drive shaft at the top by two universal joints with a telescopic coupling between them. This allows the movement of the leg from the fully retracted to the fully extended position.

The grapple fingers are independently pivoted with two contact points on each finger operating against a vertical lower cam spindle to ensure the positive movement of the fingers to open to grip or close to release the fuel assembly.

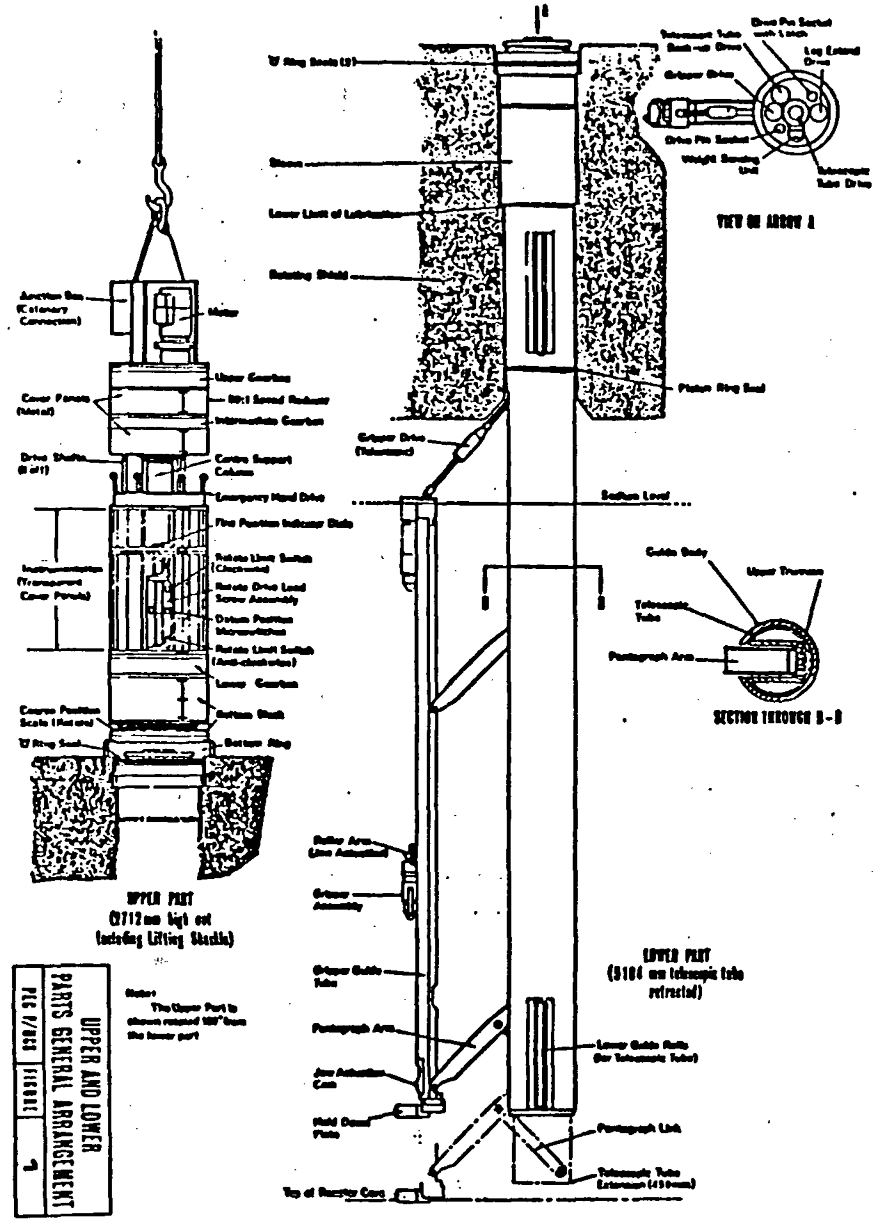
The pickup leg terminates in a holddown plate which restrains the fuel assemblies adjacent to that being lifted. The holddown plate stays in position during the full withdrawal of the fuel assembly.

The orientation of the grapple fingers to match the core assembly handling socket is controlled by a gear drive at the top of the grapple carriage. The power gear has a square hole that allows the square driveshaft to pass through. The shaft is supported in bearings at either end of the pickup leg.

When the holddown shoe at the base of the pickup leg is positioned on the core, the grapple is lowered until the sensor ring around the gripper neck is pushed upward by the core assembly head. The ring transmits the change in position to the instrumentation in the drive unit, indicating gripper engagement with the core assembly handling socket.



PFR Refueling Machine



PEC Refueling Machine

Figure F5.7-1

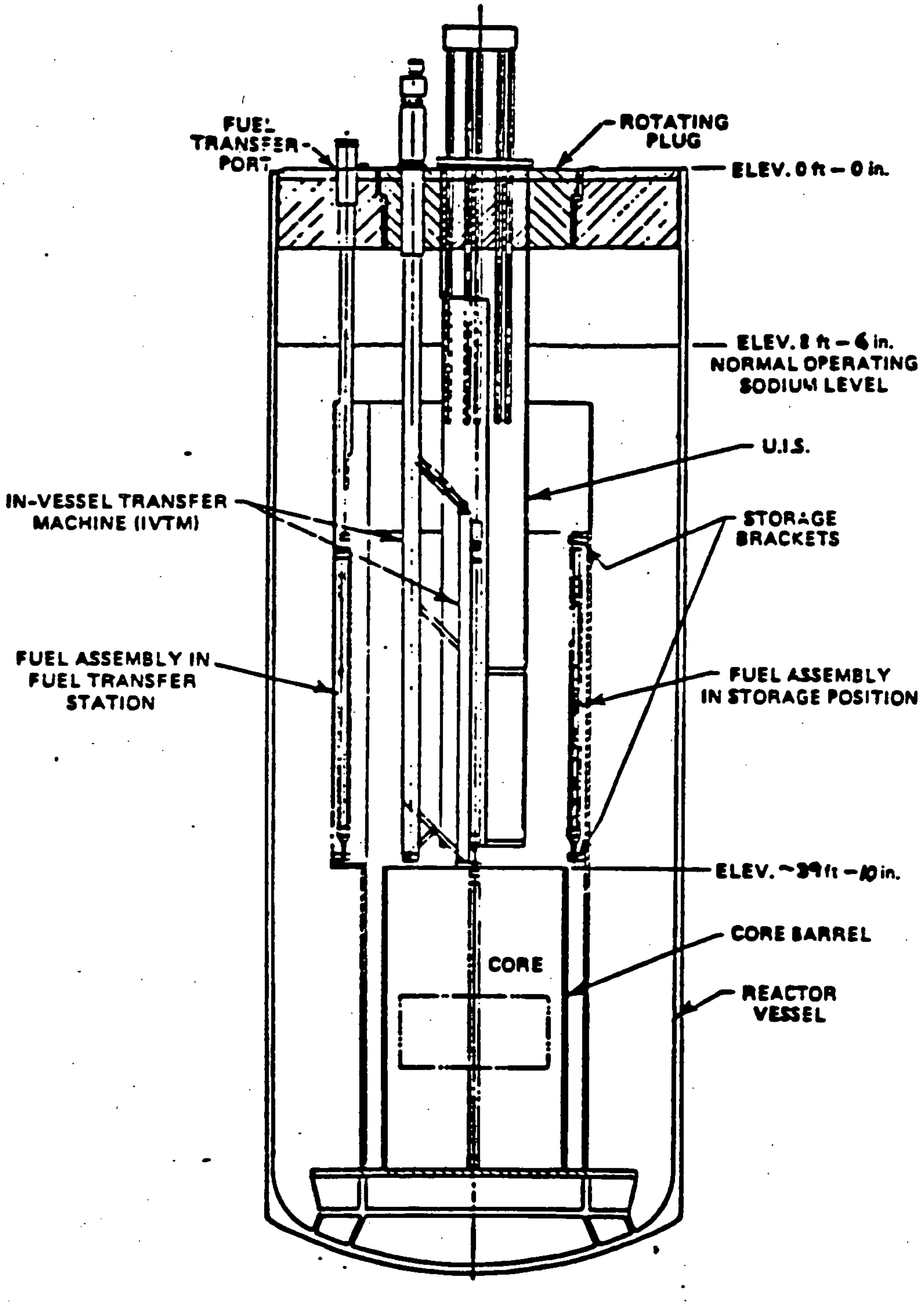


Figure F5.7-2 IN-VESSEL TRANSFER MACHINE

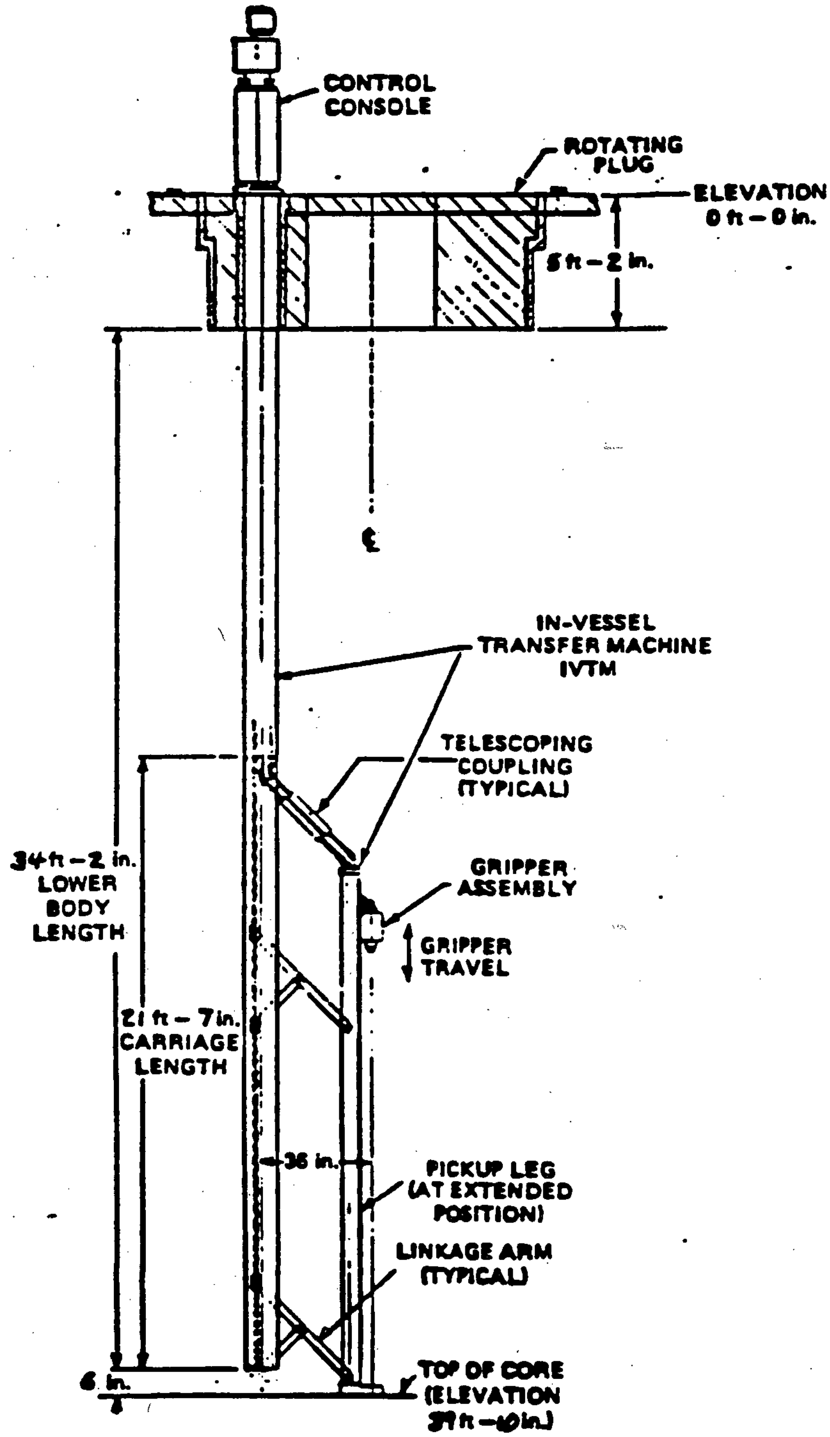


Figure F5.7-3 IN-VESSEL TRANSFER MACHINE

5.8 Comment

Vertical restraint for the assemblies is performed by the core restraint ring expanding to contact the TLP and the nosepiece receptacles. The thermal expansion of the ring and the TLP interassembly contact is difficult to predict. It seems only the nosepiece can take credit for this. Can the nosepiece alone perform the vertical restraint?

Response

There is always a net downward force on each core assembly. The various vertical forces acting on an assembly are the following:

Downward directed:
- Assembly weight (gravity)
- Hydraulic holddown (nosepiece pressure difference)
- Mechanical holddown (locking ring)

Upward directed:
- Buoyancy
- Hydraulic force (friction and pressure drop within assembly)
- Seismic (during seismic event)

At full flow conditions, the hydraulic holddown in combination with the assembly weight is sufficient to balance the buoyancy and hydraulic force to produce a net downward force on the fuel assembly equal to approximately 80 percent of the assembly weight.

A backup mechanical holddown is also provided at the assembly nosepiece/core support structure interface. The assemblies are locked into the assembly receptacles in the core support structure by expansion of the seal rings past conic lands in the receptacles. The conic lands have a 30° taper. The force required to slide th seal ring up the taper provides an additional holddown force of 100 lbf.

Considering only the hydraulic holddown force (i.e., neglecting the mechanical holddown), the core is expected, at full flow conditions, to follow the vertical seismic motions of the core support structure with accelerations up to 0.8g without lifting off the core support plate. The seismic capability improves with reduced flow until at zero flow 1.0g vertical acceleration is required to lift the assembly off the core support plate (again neglecting the mechanical holddown provided by the locking rings).

5.9 Comment

Has the area below the lower grid plate been designed to disperse molten core enough to prevent recriticality and survive a high temperature environment for long cooldown? Have you considered poisoning the bottom of the vessel to preclude a recriticality?

Response

No special features have been included in the design to disperse and increase coolability of core debris in case of an extremely unlikely core melt accident. Also, no poisoning of any portions of the lower vessel structures has been included. The preliminary risk assessment has considered such a failure sequence.

5.10 Comment

When are the thermal loads going to be factored into the mechanical stress loads for the structure to verify that no stress levels exist beyond the design limits?

5.10 Response

Thermal loads have been included in the PRISM structrural design evaluation. Three operating conditions have been evaluated:

i. Steady-state operation: gross temperature gradients
thermal striping

ii. Thermal transients: thermal shock
thermal stratification

iii. Extended RVACS operation with loss of heat sink.

The temperature distributions in the PRISM pool for these evaluations were obtained from three-dimensional COMMIX thermal-hydraulic analyses, and supplementary analyses using a lumped-mass formulation. These temperature distributions were used as boundary conditions for detailed finite element heat transfer analyses for the reactor vessel and the thermal liner during steady-state and transient operations. The pool analysis results were also used, together with the initial metal core design core exit coolant temperature predictions, to estimate the thermal gradients and thermal striping loads on the internal components. The thermal stresses in the vessel and the internal components were calculated from these estimated temperature distributions using hand-book formulas for stresses in cylinders and flat plates. The calculated temperatures are shown in Table F5.10-1 and the thermal stress estimates are shown in Tables F5.10-2 and F5.10-3.

The thermal stresses were combined with the stresses from gravity, pressure and OBE loads and compared with the ASME Code Service Level A/B stress limits to assess the adequacy of the design. The results of the comparison are shown in Tables F5.10-4 through F5.10-7.

The comparison between the stresses from gross temperature gradients combined with the gravity, pressure and OBE stresses and the stress limits for level A/B operation in Table F5.10-4 shows substantial design margins. The smallest design margin estimated for the elastically calculated thermal stresses is 0.25 at the sodium free surface in the vessel against the ASME Code stress limit for

5.10 <u>Response</u> (Continued)

shakedown to elastic action. However, the secondary stresses do not challenge the structural integrity of the vessel except through fatigue failure against which the calculated stresses have very large design margin (more than 10^6 allowed cycles against less than 2000 expected scram cycles from all the events in the duty cycle as shown in Table F5.10-5). Thus, the only consequence of exceeding the code-specified design margins would be ratcheting outwards of the reactor vessel. Considering the low temperature and relatively small number of stress cycles under consideration, this growth will be insignificant even if the secondary stress design margins are exceeded because of revised thermal-hydraulic environment estimates.

Acceptable design margins are indicated by the comparison of the calculated and allowable striping stresses and strains shown in Table F5.10-6. Actually, the results indicate that the bottom I-718 liner can be replaced by a SS316 liner. However, the stress estimates were based indirectly on a limited amount of test data for non-prototypic test configuration. Therefore, the I-718 liner is currently retained.

Table F5.10-7 shows the design margins for extended RVACS operation without heat sink. The mode of failure of concern during this operation is that of creep rupture under the sustained stresses from gravity and pressure. The calculated stresses for the vessel from these loads are 1620 psi at the sodium surface and 4270 psi at the core support attachment. The largest stress calculated for the whole structure is 9440 psi predicted at the core support attachment to the inlet plenum. This stress includes the effect of a 120 psi inlet plenum pressure which would not be acting under the extended RVACS operation condition and thus is conservative. During extended RVACS operation, the high stress regions (9440 psi-core support and 4270 psi-vessel/core support attachment) will operate at about 100°F below the upper region of vessel where the stress will be low (1620 psi). The current design specifications limit the temperatures in the high temperature, low stress region of the vessel to 1200°F for level C operation and to 1300°F for level D operation. The code-allowable life indicates that the RVACS operation can be extended to very long times at these temperatures and stress levels.

TABLE F5.10-1 OPERATING TEMPERATURES, °F

structure id	component	steady state	long-term RVACS cooling-service level C	long-term RVACS cooling-service level D	striping range
1-4	RV bottom head	610	.	1070	-
5	RV shell lower elevations	610	.	1070	-
	sodium surface	790	1090	1250	-
	cover gas	590	.	.	-
	flange support	250	350	550	-
6-10	inlet plenum plates	610	.	1090	-
23	cylinder	610	.	1090	-
11	shielding support plate	610	.	1090	-
12	flow guide	610	.	1090	-
13	support cylinder in-core	610	.	1090	830-882
15-17	(plenum wall) out-of-core	890	.	1090	-
14	shielding top support plate	610	.	1090	-
18	seal plate	610	.	1090	-
19	RV liner lower end	610	.	1090	-
	annular sod surface -230"	810	.	1090	-
	pool sodium surface -98"	888	.	1090	-
20	overflow slots	700	.	1090	-
21	horizontal baffle top plate	888	.	1090	-
23	core barrel	610	.	1090	-
24	plenum plate coupling sleeves	610	.	1090	-
25-26	core support structure	610	.	1090	-
31	UIS bottom structural plate	890	.	1090	-
	1/2" SS316 liner	890	.	1090	-
	1/4" I-718 liner	890	.	1090	828-884
32	UIS cylinder bottom end	890	.	1090	828-884
	sod. surface	890	.	1090	-
	top support	250	.	550	-
33	shroud tubes	1080	.	1090	732-980
34	instrument posts	890	.	1090	828-884

notes: . in the table signifies that temperatures are not estimated.
- in the table signifies that striping levels are insignificant.

TABLE F5.10-2 PROJECTED STRIPING LEVELS AND RESULTING THERMAL STRESSES

	peak-peak(1) core exit striping potential °F	striping attenuation in-water %	peak-peak local coolant striping °F	peak-peak metal surface striping °F	wall thickness in.	striping stress (strain) amplitude psi
UIS I-718 liner	378	82	61	55	0.25	6330(.025)
UIS I-718 shroud tube	378	19	276	248	0.10	17580(.070)
UIS support structure	354	82	57	52	1.00	8352(.036)
plenum wall	354	82	57	52	0.375	7620(.033)
instrument posts	354	82	57	52	0.75	6990(.027)
		50	159	143	0.75	19210(.076)

notes:

column 1 - The striping level based on analyses of preliminary core design not yet optimized to reduce core exit thermal gradients.

2 - Striping attenuation is the ratio of the local coolant temperature and the core exit coolant temperature from water tests.

3 - Mixing in sodium assumed to be 10% better than the water test results.

4 - Striping reduction across the coolant film assumed to be 10%.

6 - Stress calculation assumed linear reduction in striping over a depth of 0.1" and complete constraint of in-plane thermal expansion.

TABLE F5.10-3 THERMAL STRESSES DUE TO GROSS THERMAL GRADIENTS

(NON-STRIPING)

Component	circumferential stress, psi	radial stress, psi	effective strain, %	comments
STEADY-STATE OPERATION				
reactor vessel	38100	15400	0.137	@-230", sodium surface
	5200	1600	0.019	@ -12", flange
	5800	-	0.024	@-230", circumferential gradient
vessel liner	8200	500	0.033	@-230", sodium surface
	29500	11200	0.106	@ -98", sodium surface
	5300	-	0.022	@-230", circumferential gradient
UIS 1" lower plate	6620	3420	0.024	
1/2" SS316 liner	9110	1610	0.035	
1/4" I-718 liner	8680	680	0.033	
support cylinder	4920	1480	0.018	@ -98", sodium surface
horizontal baffle	7560	7560	0.031	through-the-thickness
plenum wall	3780	1130	0.016	at top of the core
THERMAL TRANSIENTS				
reactor vessel	33700	33700	0.014	peak stress
vessel liner	25350	25350	0.010	peak stress
UIS 1" lower plate	11000	11000	0.047	
1/2" SS316 liner	22040	22040	0.094	
1/4" I-718 liner	25350	25350	0.100	
support cylinder	2000	600	0.008	@ -240", stratification
shroud tubes	25350	25350	0.100	0.25" wall
	13590	13590	0.054	0.10" wall
instrument posts	36200	36200	0.143	0.75" wall
EXTENDED RVACS OPERATION				
Reactor vessel	20870	6260	0.086	@ -48" level C event
	14230	4270	0.059	@ -48" level D event

TABLE F5.10-4 DESIGN MARGINS, LEVELS A AND B - stress limits

component	calculated stress, psi	allowable stress, psi	design margin
CLOSURE	9940	27000	1.7
MODULE SUPPORT	17300	27000	0.6
REACTOR VESSEL			
bottom head - lower end	2630	49000	17.6
middle part	7050	49000	5.9
upper part	13640	49000	2.6
core support attachment	13670	49000	2.6
shell - lower region	8970	49000	5.5
sodium surface	36300	45300	0.25
INLET PLENUM			
lower plate center	5010	49000	8.8
outer circle	6200	49000	6.9
upper plate center	8730	49000	4.6
shield support	3010	49000	15.3
outer circle	5130	49000	8.5
plenum cylinnder	8980	49000	4.4
FIXED SHIELDING BOTTOM PLATE	5450	49000	8.0
FLOW GUIDE	13350	49000	2.7
SUPPORT CYLINDER			
in-core section	5920	49000	7.3
top of core elevation*	6440	21900	2.4
spent-fuel support*	1930	21900	10.3
top end*	600	21900	35.5
FIXED SHIELDING TOP PLATE	1750	49000	27.0
SEAL PLATE	8060	49000	5.1
REACTOR LINER			
lower end	9840	49000	4.0
sodium surface - pool*	670	21900	31.7
BAFFLE PLATES*	7430	21900	5.6
CORE BARREL	1190	49000	40.2
CORE SUPPORT STRUCTURE	26060	30600	0.17

TABLE F5.10-4 DESIGN MARGINS, LEVELS A AND B - STRESS LIMITS

component	calculated stress, psi	allowable stress, psi	design margin
UIS cylinder top support	3280	60000	
sodium surface*	3280	23400	6.1
UIS bottom structural plate*	1440	23400	15.2

CALCULATED STRESS = P_m+P_b+Q (components below 800F)
= P_m+P_b (components above 800F - marked with '*')
= P_m+P_b (reactor closure and module support)
OBE stresses considered primary stresses.

ALLOWABLE STRESS = $3S_m$ (components below 800F)
= $1.5S_m$ (closure and module support)
= $1.5S_{mt}$ (components above 800F)

DESIGN MARGIN = (ALLOWABLE/CALCULATED) - 1

TABLE F5.10-5 DESIGN MARGINS, LEVELS A AND B - FATIGUE LIMITS

component	stress range, psi	strain, %
REACTOR VESSEL		
sodium surface	43900	0.166
SUPPORT CYLINDER		
top of core elevation	10280	0.038
spent-fuel support	1930	0.007
top end	600	0.002
REACTOR LINER		
sodium surface - pool	30010	0.085
BAFFLE PLATES	14960	0.061
UIS cylinder sodium surface	5280	0.021
UIS bottom structural plate	12440	0.053
1/2" SS316 liner	22040	0.094
1/4" I-718 liner	25350	0.100
I-718 shroud tubes	25350	0.100
I-718 instrument posts	36200	0.143

All components have large design margins against fatigue failure from OBE and thermal transients because the allowable number of cycles for the above ranges of strain is greater than 10^6.

CALCULATED STRESS = P_m+P_b+Q
OBE stresses considered primary stresses.

TABLE F5.10-6 DESIGN MARGINS - STRIPING FAILURE

	wall thickness in.	striping stress(strain) amplitude psi (%)	allowable stress-I718 strain-SS316
I-718 COMPONENTS			
UIS bottom liner	0.25	6330 (.025)	31400 psi
UIS shroud tube	0.10	17850 (.070)	in absence
instrument posts(82%)	0.75	6990 (.027)	of mean
(50%)	0.75	19210 (.076)	stress.
SS316 COMPONENTS			
UIS support structure	1.00	8352 (.036)	0.13% in
plenum wall	0.375	7620 (.033)	absence of mean stress

TABLE F5.10-7 DESIGN MARGINS, LEVEL D - RVACS - STRESS LIMITS

component	calculated stress, psi	temperature °F	allowable life, hrs
VESSEL UPPER PART	1620	1250	long
		1300	long
		1350	10^7
		1400	500000
VESSEL CORE SUPPORT ATTACHMENT	4270	1100	long
		1200	500000
		1300	20000
CORE SUPPORT STRUCTURE	9440	1100	200000
		1200	10000

CALCULATED STRESS = P_m (NORMAL OPERATION - NO THERMAL STRESS)

5.11 Comment

Three modules feed one turbine. If one module goes thru an event, will the problem propagate to all three plants? How is module isolation accomplished so the other systems are not effected (i.e., say more about the remote isolation valve)?

- o Why not have a turbine for each module and guarantee that problems will not propagate?

Response

Control analysis has shown that the start up, or even the tripping of one of the three modules in a power block does not impact the operation of the remaining modules. For example, when one of three modules trips from full power, the reduced steam flow from the tripped module results in temporary steam drum pressure reduction (in all three steam drums) of only 40 psi in the first 50 seconds. The steam drum pressure returns to its normal pressure of about 980 psi (@ 2/3 power) within the next 100 seconds as the turbines' throttle valves adjust to maintain the set point steam pressure. Steam side isolation of the tripped module is not required. If the steam and intermediate sodium systems are in tact and operating in their natural circulation mode following the trip, the tripped module will continue to supply about 10% of rated steam flow during the initial part of its cooldown to an isothermal hot standby temperature of 550°F, Thus, for this "normal scram" event there is no need to close the dual steam line isolation valves on the tripped module.

Dual automatic isolation valves are provided on the steam lines exiting each steam drum to decouple a faulted system from the other modules. The automatic steam line isolation valves are programmed to close on high steam flow, low drum level, and a SG rupture disk failure.

5.12 Comment

Have the IHTS piping gimballed bellows joints been used before? How will the reliability of the articulated joint be established?

Response

The gimball bellows joint has been used commercially before, but has not been used in sodium service. Rockwell has qualified a 12-inch diameter bellows for nuclear application and use in sodium lines. The gimballed bellows designed for PRISM will be developed and demonstrated in a supporting test program by ETEC. This program is scheduled to start in 1988 and will include fabrication and test of a full scale gimballed bellows.

5.13 Comment

How will the flare tip and hydrogen ignitor (from SWRP) still be operable after a large scale blowdown from multiple tube ruptures in the SG?

Response

Flare stacks and ignitors of similar design have been successfully tested and used at ETEC during the steam generator large leak test program.

5.14 Comment

Where are the sodium leak detectors located in the SG and what are they? What will you do if one of the leak detectors malfunctions during plant operation?

a. The measurement of hydrogen in the sodium by diffusing thru a Ni membrane seems to be such a slow process for such a fast reacting event.

b. What is the response time of the detector?

c. Is the blowdown into the SDT and the scramming of the reactor automatically or manually controlled?

Response

There are three redundant hydrogen leak detectors located on the 30-inch diameter sodium outlet line from the steam generator. These detectors monitor the main sodium stream leaving the steam generator. In addition, there are three redundant hydrogen leak detectors located on the 3-inch diameter vent line which provides a positive flow through the upper tubesheet region of the steam generator. The leak detectors measure the hydrogen concentration in the liquid sodium stream by allowing hydrogen to diffuse through a thin nickel membrane, one side of which has a high vacuum held by an ion pump. Three redundant detectors are provided for each sodium stream leaving the steam generator to accommodate detector malfunctions during operation. A minimum of two operational hydrogen meters are required at each point of measurement to ensure

5.14 <u>Response</u> (Continued)

leak detection coverage any time steam/water and sodium are present across the boundary of the steam generator tubes. The plant shall be shut down if this condition is not met. The detectors are designed to allow replacement of the diffusion membrane without removal from the loop.

The purpose of the leak detector subsystem is to alert the plant operator to the presence of small water-to-sodium leaks within the steam generator units to allow the operator to take corrective action. The response time of the detection module is about 30 seconds. The response time is defined as the time for sodium transported from the inlet of the detector to the time when the hydrogen in the sodium is sensed at the instrumentation readout.

The detection system functions in a leak size range that is substantially smaller than a major water leak and has no function in an event that leaks large amounts of water.

In the event of a major sodium/water reaction, the rupture disc burst and the high pressure in the down stream piping automatically scrams the reactor and isolates/blowdown the steam/water side of the steam generator.

5.15 <u>Comment</u>

Can the modular vessels be lifted out of the silos for silo repair if damage occurred from an earthquake?

<u>Response</u>

The reactor module, after removal of the core, sodium and other necessary components, can be lifted from the silo in the same manner as planned for replacement after 60 years.

The collector cylinder (RVACS) can also be removed so that the silo (concrete cylinder in contact with the earth) can be repaired as necessary. The silo is a reinforced concrete structure designed to withstand an SSE event with margin. Other concrete structure is used for radiation shielding and is part of the system that is on isolators and, therefore, less sensitive to an earthquake event.

5.16 <u>Comment</u>

Explain the control system related advantages to the saturated cycle.

<u>Response</u>

The module to module coupling effects are minimized in the reference saturated steam cycle design. The only cross-coupling between the individual modules occurs at the steam generator system feedwater and steam headers. The feedwater flows from the common feedwater header to the individual modules are independently controlled, maintaining the individual steam drum water level. In a saturated cycle the steam temperatures are determined by the system steam pressure. The individual module steam temperatures are essentially identical since the flows join at a common pressure, the exit steam header. Therefore no steam valving is required to add or remove a steam generator system from power operation as is required for a super heat cycle which has individual module steam temperatures varying with module power level.

The saturated cycle steam generator system with recirculation loops enhances stability. The increased steam drum volume and steam-water surface area reduces pressure perturbations. The corresponding steam temperature variations are also reduced. The large steam drum inventory reduces the speed and accuracy requirements for the feedwater flow control valves compared to that needed for a smaller inventory once through superheat cycle.

The ample steam generator heat transfer surface area acts to hold the IHTS cold leg sodium temperature near the water-side saturation temperature. Water side pressure and temperature are held essentially constant over the operating power range. This maintains the IHTS cold sodium temperature essentially constant over the operating power range.

Constant primary and intermediate loop sodium flows are maintained in each module over the entire power operating range. This further simplifies plant control.

Module cooling following shutdown continues to utilize the general steam system heat removal path. Should the steam system be unavailable, the Auxiliary Cooling System (ACS) using simple damper/fans air cooling controls and the RVACS requiring no active control would be utilized for shutdown cooling.

5.17 Comment

Explain the ACS system and indicate its expected reliability.

Response

The Auxiliary Cooling System (ACS) is used for decay heat removal when the normal heat removal path to the condenser heat sink is not available. The ACS is a non safety-related system and is used to supplement the RVACS and reduce the time required to cool the reactor system and IHTS following a reactor shutdown when the steam system is not available. The ACS consists of a shroud around the steam generator which provides a six-inch wide cooling annulus. The steam generator is cooled by natural circulation of air over the exterior shell surface upon actuation of the inlet and outlet louvers. Inlet air is taken from the steam generator building and exhausted through a stack to the atmosphere. The ACS has a rated heat removal capacity of 2 MWt.

The ACS only requires the opening of the inlet and outlet louvers to place the system in operation. The louvers are open by diverse actuation mechanisms; a) remote, pneumatic, failsafe operators and b) local, manual operator override. The louvers will be periodically tested during plant operation. Once the louvers are open, the system operation is passive and inherent in that it only depends on natural circulation of air through the shroud/shell annulus. The ACS is, therefore, virtually failproof with a reliability approaching one.

5.18 Comment

Has the effect of acid rain (or air quality problems) been factored into the lifetime of the containment vessel?

Response

The containment vessel material (2-1/4 Cr-1Mo) is not susceptible to environmental effects. The operating temperature of the containment vessel is everywhere higher than the boiling point of water for all operating conditions. This will reduce or eliminate the potential for deposits on the vessel.

The produced containment vessel will be uniformly coated with an oxide coating by a subcritical (below the Ac_1 temperature) heat treatment in Air, 1340°F for 1 hour minimum. This heat treatment produces a tenacious and inert oxide coating which will resist environmental problems such as further oxidation or corrosion pitting.

5.19 Comment

How much radiation will be exhausted from the RVACS system?

a. Activated fragments from the air and the air collector passage will be sent up the stack. What is the health hazards to surrounding areas?

b. How will fouling be removed from the cold air downcomer and the hot air riser if the need arises (or any maintenance problem)?

5.19 Comment (Continued)

c. If the monitored exit air from the RVACS deviates from the expected will the reactor be shutdown? What criteria will you use for the expected range of temperatures?

Response

The activation in the air effluent consists primarily of Ar-41 and N-16 and any activated contaminants that may be exhausted from the RVACS with the air. The contribution of C-14 is small compared to the activity resulting from these two isotopes. The calculated activity concentrations for Ar-41 and N-16 are respectively 2 X 10^{-17} μCi/cc (air) and 9 X 10^{-13} μCi/cc (air). The corresponding 10CFR20 activity concentration limits are 4 X 10^{-11} μCi/cc (air) and 3 X 10^{-11} μCi/cc (air) for an unrestricted area.

Since Co-60 is commonly present in the earth's crust, an estimate was made of the maximum allowable concentration of activated ordinary dust in the RVACS exhaust air so as not to exceed the 10CFR20 Maximum Permissible Concentration (MPC) limit for Co-60 (3 X 10^{-13} μCi/cc (air) for an unrestricted area). For the very conservative irradiation time of one hour, a maximum of 13g of dust per cc of air can be exhausted. This value exceeds the average density of ordinary dust by at least a factor of five. Thus, the in-vessel radial shielding will limit activation of the RVACS effluent to acceptable values.

RVACS thermal performance analyses have shown that postulated fouling of the RVACS heat transfer surfaces has a negligible effect on the RVACS heat removal capability. Thus, fouling on the heat transfer surfaces will not need to be removed during the life of the plant. The RVACS air passages will be inspected at periodic intervals using remote viewing equipment. If inspection shows that dust or debris have accumulated at specific locations, e.g., at bottom of the reactor cavity where the air stream turns, to the point where the air flow rate is reduced, steps will have to be taken to remove the flow restriction. However, the volume of debris that can be accommodated is very large and the need for its removal is not anticipated during the life of the plant. Should such a need arise, it would be possible to use a remotely controlled vacuum cleaner in combination with the remote viewing equipment. The thermal performance of RVACS is continuously monitored by measuring the air flow temperature and rate in each air outlet stack.

No radiation monitors are placed in the RVACS exhaust ducts since radiation and sodium aerosol detectors in the Reactor Head Access Area will alarm if cover gas leaks through the reactor closure, and leaks from the reactor vessel will be detected by aerosol detectors monitoring the annulus between the two vessels.

5.20 Comment

Where are the core outlet temperature detectors installed and what are their response time? Any automatic scram associated with them? How many must operate for the plant to stay on line?

5.20 Response

The upper internal structure (UIS) has 20 drywells routed from the top of the rotatable plug to the region directly above the reactor core. These pass through and are supported by the UIS. Each SS-316 drywell is 0.5 inches in diameter and will carry multiple thermocouples. The lower end of the drywells, which are located approximately 1.5 feet above the core outlet, are contained in heavy Inconel Alloy 718 forgings that provide structural support and thermal protection. The posts are designed for mechanical loads resulting from any sodium flow induced vibration. The Inconel Alloy 718 material is specified to guard against thermal fatigue due to the steady state thermal striping and the thermal shock arising during scram transients.

The drywells are evenly distributed over the core outlet plane so as to provide information on the various core regions. Above this plane they are routed so as to allow them to exit the reactor through a single port, thus minimizing the number of required plug penetrations. Between the port and the instrument post the drywells are enclosed in conduits and ducts that facilitate their routing and provide mechanical protection.

The overall response time of these thermocouples is on the order of 10 to 20 seconds.

The above core thermocouples are used for reactor control, to provide core outlet temperature information to the operator, and are used to scram the reactor.

The minimum number of thermocouple locations that must be available for the reactor to stay on line is estimated at 15 out of the 20. It is very unlikely that the reactor will be off line due to the unavailability of these thermocouples. Redundancy is provided in two ways: (1) redundant drywell locations and; (2) multiple thermocouples per drywell. Each drywell will have multiple thermocouples. If one of the units within a drywell fails, the circuit is simply switched to the in-place spare. If all the thermocouples in the same drywell fail, sufficient temperatures coverage is provided by the thermocouples in the remaining drywells.

5.21** Comment

Please provide the equations describing the EM pump operating characteristics.

Response

The equations are given in Section 5.4.3.3, EM Pump performance characteristics.

**Comments for further consideration.

5.22** Comment

It is noted that the EM pump cables are protected from the high temperature sodium by gas filled tubes. What is the maximum time/temp transient the cables can withstand? How many cables can fail before adequate coolant flow is lost?

Response

The power leads are planned to be of the same insulation used on the pump coils. This insulation can withstand thousands of hours at 1400°F without failure although its life will then be shortened in such service. The pumps are not planned to be operated at RVACS conditions so the voltage stress is not present making the condition less severe than what has been tested.

Failure of one cable will cause pump failure. Three remaining pumps will provide adequate coolant flow until shutdown is achieved.

5.23** Comment

Since no active system is provided for adding argon gas to the CV/RV annulus, how will gas leakage be determined? How much leakage is permissible during operation?

Response

During all phases of reactor operation, after startup, the CV/RV annulus gas is a positive pressure (14 psig). Leakage of gas from the annulus will be detectable on the pressure sensors installed therein.

An indication of leakage would be followed by detection examination of the only likely leak locations, the welded circumferential omega seal between the closure and the containment vessel flange and the welded covers over the closure-to-containment vessel flange connection bolts. (A leak at these locations is easily repaired.) Loss of pressure in the annulus would also be followed by gas sampling and analyzing for either helium from the reactor cover gas or air from the outside.

Theoretically, leakage from the annulus could be tolerated as long as the pressure remained positive to indicate continued existence of an inert atmosphere in case the reactor vessel should leak. In practice, however, leakage in excess of 0.1% volume/day would be permissible.

Argon gas leakage from the CV/RV annulus is detailed in Sections 6.2.1.3 Design Evaluation and 6.2.1.4 Testing and Inspection.

5.24** Comment

What are the minimum spent fuel assembly cooling requirements before removal from the reactor vessel?

**Comments for further consideration.

5.24 Response**

The fuel transfer cask (FTC) is designed to handle 3 spent fuel elements after 20 months in-vessel storage and maintain a fuel pin cladding temperature below a 1200°F limit. The FTC is capable of removing about 5.6 Kwt decay heat without exceeding the spent core element temperature limit. The decay heat for spent fuel elements after 20 months in-vessel storage is down to 1.33 Kwt each. Transferring three assemblies in the FTC will amount to 4.0 Kwt decay heat.

5.25 Comment**

How many seismic isolators are necessary to carry the vertical load (static) of the facility?

Response

Twelve (12) seismic isolators, assuming uniform loading, will support the static vertical load of the facility without exceeding their design loading. The seismic isolator tests, to be performed at ETEC in 1988, are expected to show that the design load is a factor of two or three below the vertical load required to fail an isolator.

5.26 Comment**

Thermal Stresses (Strains)

It is not clear that the treatment of thermal stresses (strains) has been covered adequately or completely. On p. 5.1-5 it is stated that thermal transients will be benign with respect to the core support because of the distance to the source(s) which implies that elsewhere such transients may be malignant. At the least, elaboration would help. On p. 5.2-11 it is stated: "...thermal stresses in the reactor vessel are acceptable." To support that conclusion and other statements made in the same paragraph:

a. tabulate the results of stress analyses with comparison to the experimental material behavior under relevant conditions of temperature, stress ratio, frequency, environment and material condition;
b. ensure that any effects or erosion (or loss of metal by any other mechanism) are included in the analyses (see Item 5.27**, below).

At the end of the first paragraph, p. 5.2-12, a similar statement is made about liner thermal stresses which, also, should be supported with results. The first paragraph on p. 5.2-13 clearly says that the thermal analyses are incomplete; that work must be completed satisfactorily (see Item 5.30**, below). On p. 5.3-10 the second paragraph says, in part: "...are protected...from the high sodium flows and the rapid temperature changes that will occur in the hot pool." Those words imply that the thermal strains may well be high enough to be of serious concern. Similarly, on p. 5.3-22, the text (at the bottom of the page) says an "...outer liner is used to insulate the structure against rapid temperatures changes occurring during scram transients," again implying that large thermal strains will occur. On the next page, 5.3-23, in the second

* Comments requiring textual improvement.
**Comments for further consideration.

5.26** Comment (Continued)

paragraph again it is stated that the thermal stress analyses are not yet detailed because the thermal environment has not been sufficiently characterized. Likewise, on p. 5.3-25, the first two paragraphs under Thermal Loads repeat the statement that more needs to be done and it appears conclusions are based as much on expectations as on results. Page 5.3-26, in the first two paragraphs under UIS Lower Assembly also state detailed work has not been done. Finally, p. 5.3-27, under the title UIS Support Cylinder and the sentence preceding that title, also says that thermal stresses have not been evaluated in detail but are expected to be low. In the face of all the work yet to be done, what confidence is there in the adequacy of the design to withstand the thermal stresses and strains and provide for safe operation?

5.26** Response

Refer to the response to Comment 5.10.

5.27** Comment - Liquid Na Effects

Erosion - Corrosion

Long-term exposure to liquid sodium may result in material degradation beyond that discussed in the PSID. Consider the following: page 5.2-5 (bottom paragraph), page 5.2-7 (top paragraph), page 5.3-3 ("...conditions to be taken into account...shall include...the effects of the sodium environment."), page 5.3-5 (item (1) under paragraph 5.3.1.3), page 5.4.1.9 and page 5.4-26 (bottom paragraph). The potential for erosion, corrosion or erosion-corrosion by the following liquid sodium should be discussed more fully. It is suggested that:

a. Data are required regarding the erosion of type 316 stainless steel under tensile strain by liquid sodium at the temperatures and flow rates which will obtain PRISM and the results discussed in or around page 5.2-5.

b. The need to include an allowance for erosion-corrosion be included in Section 5.3.1.1, possibly as the item following either (11) or (12), page 5.3-2.

c. Include the need to address liquid sodium erosion-corrosion effects in Section 5.3.1.2, pp. 5.3-2, -3.

d. Further discussion should be included in or around Item (2), Section 5.4.1.9, page 5.4-6.

e. Address corrosion by liquid sodium under Duct Materials on page 5.4-26.

Streaming

The potential for in homogeneous flow of liquid sodium was noted on pages 5.3-13 (top paragraph) and 5.3-26 (bottom paragraph). Even if general, uniform, flow

**Comments for further consideration.

5.27** <u>Comment</u> **(Continued)**

would not be expected to result in erosion, corrosion or erosion-corrosion, locally intense streams could be cause for concern, especially if they entail local thermal variations, and the condition should be addressed.

<u>Response</u>

The sodium exposure is not expected to affect the performance of the reactor structures at the relatively low operating temperatures. The exposure effects can be taken into account by providing a thickness allowance and by modifying the material properties in the design evaluation as appropriate.

<u>Erosion-Corrosion</u>

Erosion-corrosion may be accounted for by reducing the load bearing thickness of a component when performing stress analysis. The wall-thinning rates for 304SS and 316SS as given in the Nuclear Systems Materials Handbook (NSMH, TID 26666, ORNL, Oak Ridge, TN) are shown in Figure F27-1. The rates are given as a function of oxygen content, sodium velocity and sodium temperature all of which tend to increase the corrosion rate. The corrosion rates for Alloy 718 are comparable to those shown for the austenitic steels in Figure F27-1. The rates in the figure indicate that the corrosion effects for the PRISM reactor structures will be insignificant (<0.005" in 60 years) in view of the low oxygen concentration (2 ppm) in the reactor sodium, operating temperatures below 900°F, and low fluid velocities.

<u>Property Changes</u>

Exposure to flowing sodium may also produce changes in the alloy constituents with corresponding changes in the material properties. These changes are divided into surface effects resulting from transfer of metallic elements and interstitial transfer effects which affect the material to a greater depth.

<u>Surface Effects</u>: Depending on the operating temperatures, liquid sodium may cause transfer of certain metallic elements from the hotter to cooler regions of the system. This has insignificant effect on the material short-term tensile properties. The creep-rupture times, on the other hand, are reduced by presence of sodium (helium). In creep tests ranging up to rupture times of 10,000 hours, sodium was observed to produce a constant decrease in rupture strength compared to air as indicated in Figure F27-2. The data in the figure indicate that this reduction will not affect the PRISM reactor structures because of the low operating temperatures (<900°F).

<u>Interstitial Transfer Effects</u>: In an all-austenitic system, sodium transfers interstitial carbon and nitrogen from hotter to the cooler regions. In austenitic/ferritic systems, the transfer is generally from ferritic to austenitic parts of the system. However, some decarburization in the austenitic part of the system may occur even in this case because of the interstitial transfers in the austenitic range of the system. Interstitial transfer leads to weakening in the decarburized and denitrided areas and to strengthening in the carburized and nitrided regions. The surface strengthening due to carburization also increases

**Comments for further consideration.

the elastic strain range and therefore increases the fatigue lives in the case of strain-controlled fatigue loads. However, this is partially offset by enhanced crack nucleation at carbide precipitates.

The approach developed for the CRBR design to account for the interstitial transfer was to calculate the local interstitial levels in a component as a function of exposure time and temperature, to calculate the effect of the change in the interstitial level on material properties, and to use the modified material properties in stress analysis and design evaluation. Preliminary calculations using this approach indicate insignificant effect on the PRISM reactor structures as indicated below. The discussion is for carbon transfer effects in SS316. The strength properties are less sensitive to nitrogen and diffusion in SS304 is slower or comparable. Therefore the conclusions are applicable to all the PRISM reactor structures.

The carbon diffusion coefficients for SS316 extracted from NSMH are shown in Figure F27-3. Using the following approximate formula from NSMH to calculate the diffusion depth for 60 years exposure at 900°F:

$$
\begin{aligned}
d &= 2.(\text{diffusion coefficient x exposure time})^{1/2} \\
&= 2.(2\text{x}10^{-13} \times 60 \times 365 \times 24 \times 3600)^{1/2} \\
&= 0.0389''
\end{aligned}
$$

carbon diffusion will affect less than 0.04" of the thickness. For a linear carbon gradient in the affected thickness, less than 0.02" of the thickness may be assumed to have carbon content in equilibrium with the sodium system carbon potential and remaining thickness may be assumed to be unaffected. For a component exposed on both sides, the thickness with equilibrium carbon concentration will be double, but still less than 0.04". With the PRISM component thicknesses equal to or greater than 1", this thickness amounts to less than 4%. Even with a complete loss of strength in this material, the component would retain more than 96% of the short-term tensile strength. Actually, carbon diffusion will be slower because of the lower temperature, and the material will not lose its strength completely even when denuded of carbon. Therefore, change in the short-term tensile capability will not be significant.

The effects of interstitial diffusion on the SS316 stress-rupture properties will be insignificant at the PRISM component temperatures (<900°F) as indicated in Figure F27-4.

Loss of interstitial concentration has been observed to increase the fatigue life. Absence of surface oxidation under sodium also increases the fatigue life. However, these effects are not sufficiently quantified for their use in design.

Streaming

The general sodium velocity in the primary circuit is less than 5 ft/second. This velocity may locally increase up to 20-30 ft/second at the core exit. As indicated in Figure F27-1, and discussed in a previous paragraph, the corrosion

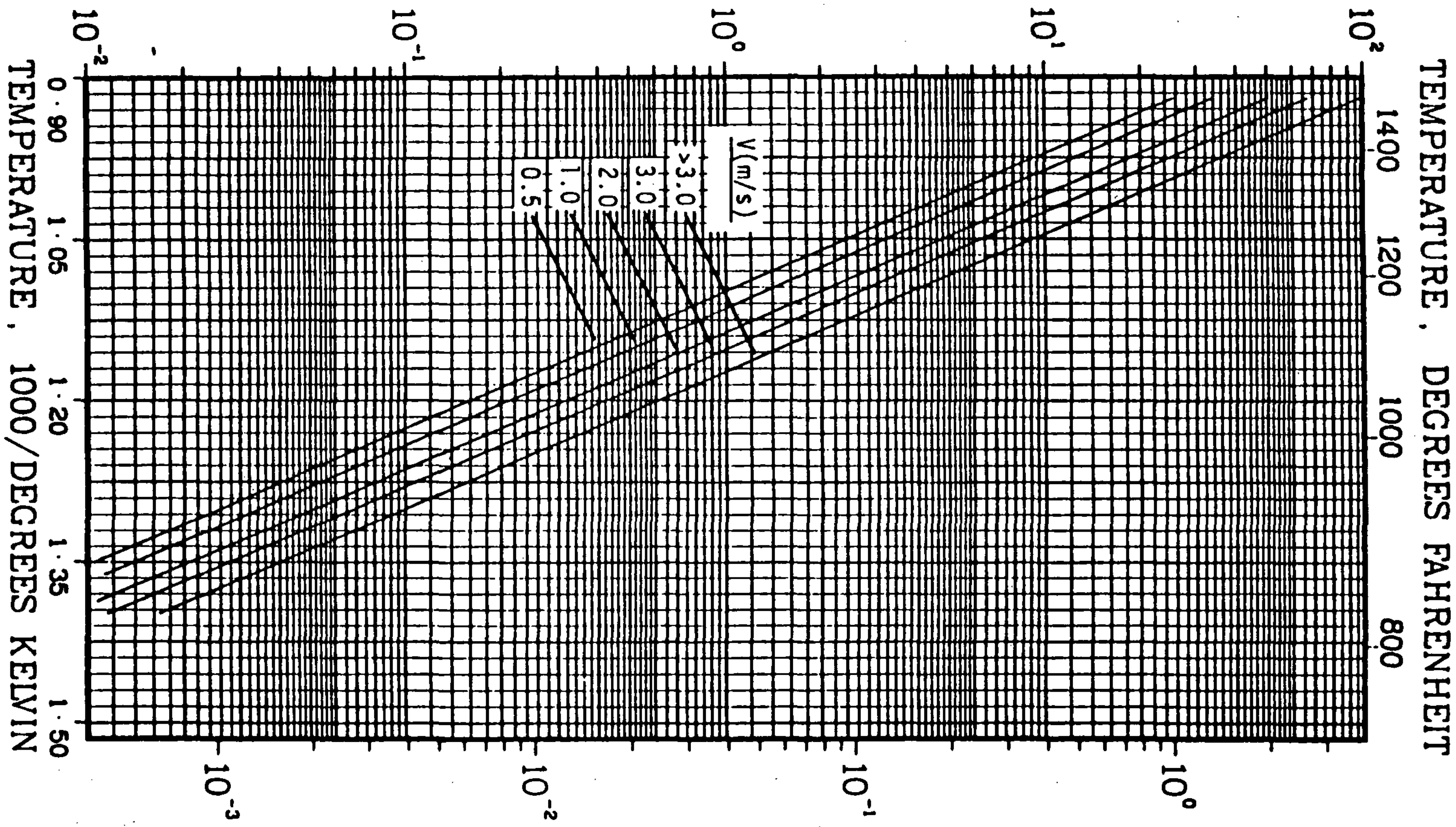
**Comments for further consideration.

from such velocities will be acceptably small due to the low oxygen contents and the low temperatures of the components exposed to the flowing sodium.

The thermal striping effects of sodium streaming are discussed in the response to Comment 5.10.

**Comments for further consideration.

CORROSION RATE, MICROMETERS/YEAR-PPM OXYGEN



CORROSION RATE, MILS/YEAR–PPM OXYGEN

APPLICABLE PRODUCT FORMS: *All forms and heat treatments.*

NOTES: *Read the notes at the end of the tables before using these curves.*

Figure F27-1 Corrosi... ..n Liquid Sodium

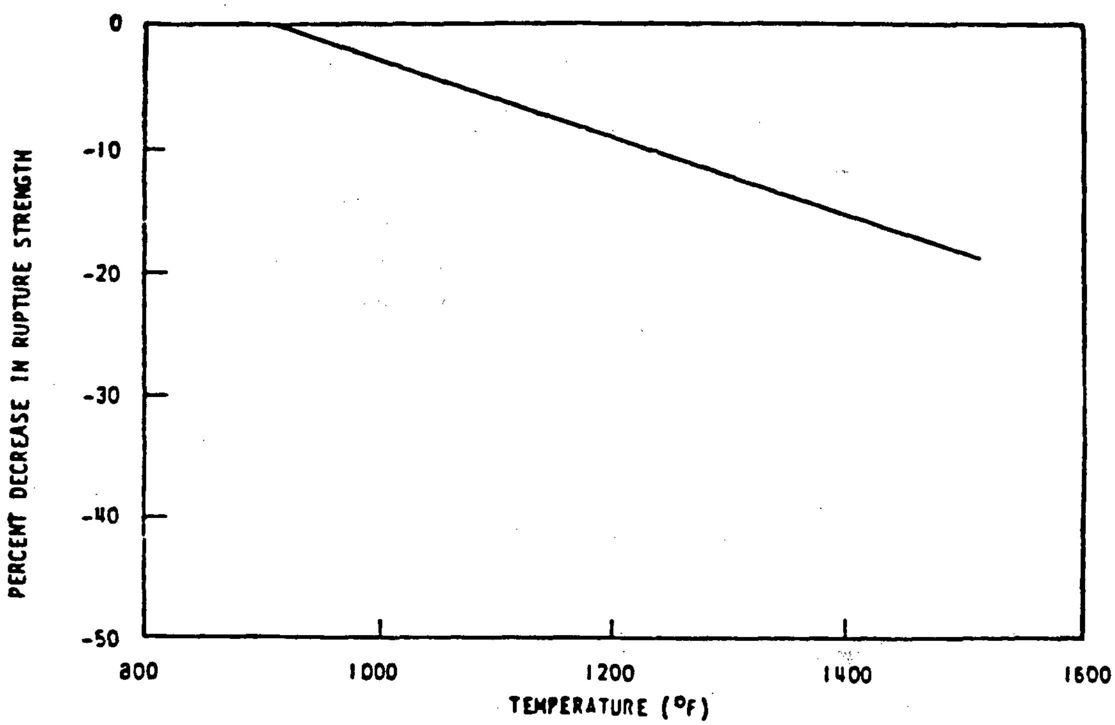


FIGURE F27-2 DESIGN FACTORS FOR ESTIMATING THE DECREASE IN RUPTURE STRENGTH OF SS316 FOR SURFACE INTERACTION WITH SODIUM

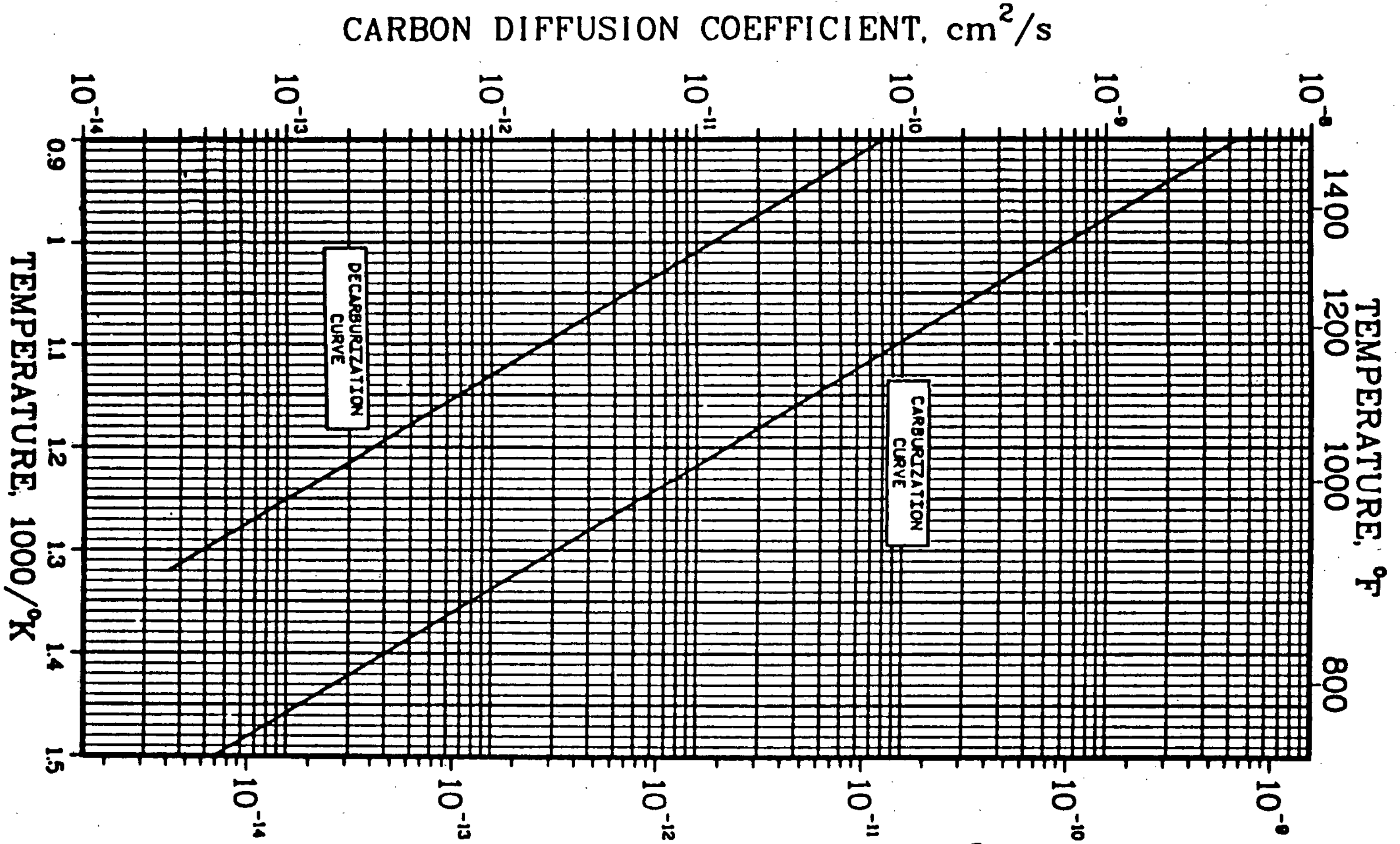


Figure F27-3 CARBON DIFFUSION COEFFICIENT, in^2/s

PROPERTY CODE 4202

APPLICABLE PRODUCT FORMS
NOTES *See page 1.1.*

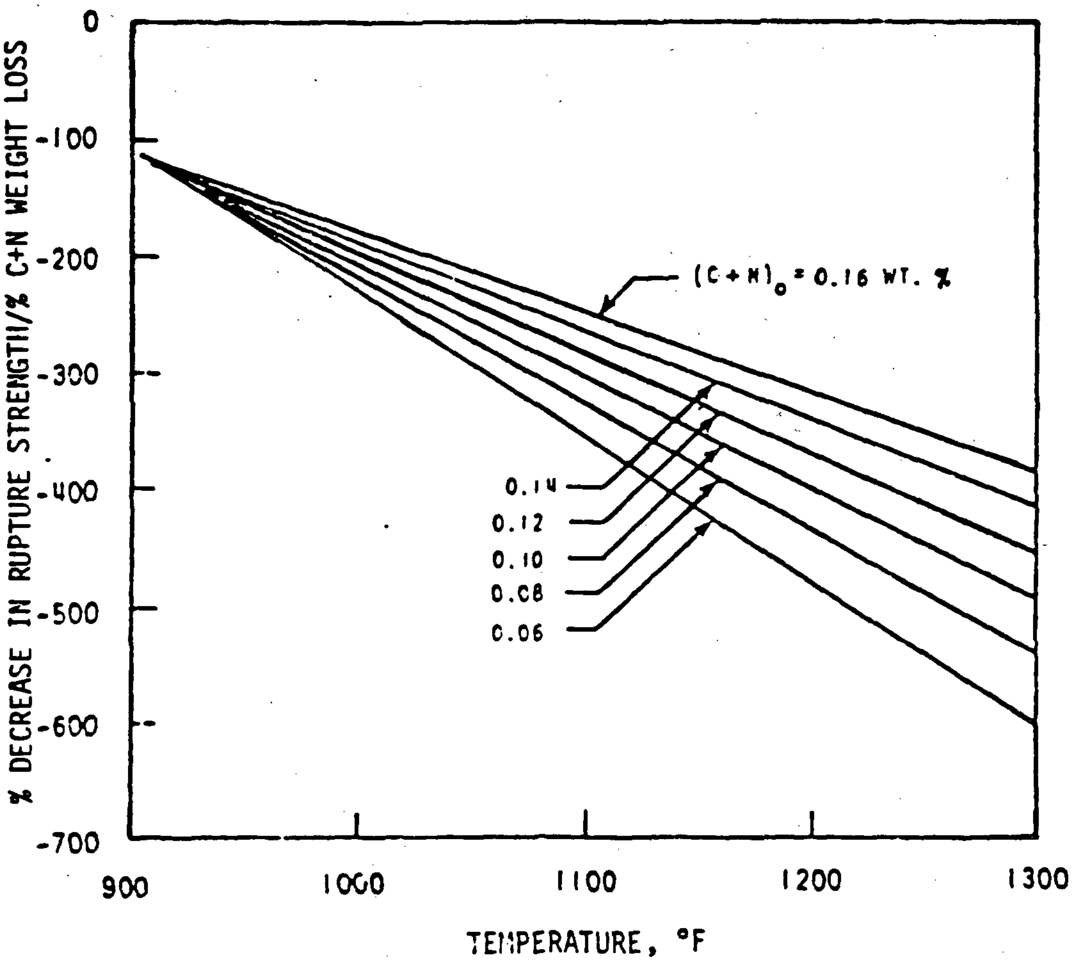


FIGURE F27-4 DEGRADATION IN RUPTURE STRENGTH DUE TO LOSS OF INTERSTITIAL CONCENTRATION

5.28** <u>Comment</u> - <u>Inspection/QC</u>

Does the rotatable plug design, Figures 5.2-3 and -4, meet the requirement of the ASME Code, Section III? If so, specify where and how as part of "Design <u>Criteria</u>", page 5.2-4. Regarding Section 5.2.2.3, page 5.2-7 ff, how will the closure head, with such a complicated design, meet the ASME Code, Section XI, ISI requirements? A discussion of this point should be added to Section 5.2.2.3.

If 10CFR50, Appendix B, requirements will be met why not say so explicitly, with some discussion, in Section 5.4.1.10?

5.28** <u>Response</u>

The rotatable plug is considered to be a part of the reactor closure and therefore subject to the same design requirements and design criteria. Section 5.2.1.2 is revised to clarify this point.

The reactor closure design is not complicated. It is a flat head with penetrations, a configuration commonly recognized by the Code. Its single structural plate boundary is particularly advantageous for ISI by leakage monitoring in the head access area, which is in specific compliance with Section XI Division 3, of the ASME Code. See the change made to Section 5.2.2.3.

Section 5.4.1.10 Quality Assurance Requirements is not consistent with the Chapter 5 format and is deleted. Quality Assurance Requirements are contained in Chapter 17.

5.29** <u>Comment</u> <u>MATERIALS</u>

a. <u>Comment</u> - <u>Bellows</u>

The inclusion of bellows in the PRISM design was mentioned on pages 5.2-11, 5.5-1 and 5.5-19. The basic function of articulating bellows imposes cyclic strains on the metal from which they are made. The related discussions should be expanded to show how such potential failures will be avoided. Include consideration of the potential for change in fatigue cracking resistance as a consequence of exposure to radiation and temperature (see Item d, below).

<u>Response</u>

As noted on page 5.5-19, gimballed bellows are used in the IHTS piping to accommodate thermal expansion and seismic anchor displacement. These bellows are not exposed to any appreciable radiation as they are located in the secondary sodium (IHTS) system. The gimballed bellows designed for the IHTS will be developed and demonstrated in a supporting test program by ETEC. This program is scheduled to start in 1988 and will include fabrication and test of a full size gimballed bellows.

**Comments for further consideration.

b. Comment - Rupture Discs

The inclusion of rupture discs in the PRISM design was mentioned on pages 5.5-6 -8 and -9. The use of rupture discs involves careful attention to detail to ensure that they will rupture when it is necessary, that they do so and that they will not rupture when they should not. Therefore, in order for us to review this design facet, more detail is needed, most likely as part of Item (4) at Section 5.5.1.2.1, page 5.5-8.

Response

The rupture disc will be designed to burst within ±5% of the required pressure (300 psig) and at the system normal operating temperature (550°F). Sample rupture discs from each lot shall be burst tested in accordance with the requirements of the ASME Code. It should be noted that an accurate burst pressure is not required, as the setting of 300 psig is well above the system operating pressure (<100 psig) and well below the IHTS design pressure (1000 psig).

Dual rupture discs (two discs in series) are used to prevent inadvertent system dump in the event of a rupture disc failure. If the upstream disc fails, the second disc will maintain system integrity and a sensor between the discs will alarm and allow the operator to take corrective action.

c. Comment - Sensitization

On pages 5.2-4 (Item (2), Section 5.2.1.3), 5.3-5 (Item (4), Section 5.3.1.3) and 5.4-6 (Item 4), Section 5.4.1.9), the text speaks of doing appropriate things to minimize sensitization. In all three places, the discussion, taken in context, deals with more than stainless steel materials yet sensitization usually is not considered with respect to ferritic steels (such as the Cr-Mo steel of the containment vessel). If the discussions about sensitization were, indeed, intended to apply to stainless (austenitic) steel only, say so and divorce that part of the text from other materials; if not, explain what is intended. Also, will sensitization, if it is present, play a role in the relation of stainless steel to the liquid sodium environment? Further, will the guidance of NUREG-0313, Revision 2, be followed regarding the choice of stainless steels? If so, say so (and why and how); if not, explain why not.

Response

Text dealing with sensitization relates only to the austenitic stainless steels since these steels are the only ones susceptible to intergranular attack when exposed to moisture. Of prime concern then is the period during fabrication, storage, and installation prior to plant startup. Although proper precuations will be taken to prevent/minimize sensitizing these steels prior to plant startup they will become sensitized during plant operation since the primary system will operate in the sensitizing temperature regime for these steels, >700°F. Effects due to sensitization during service are neglected since localized attack does not occur in these steels when exposed to controlled sodium environments.

**Comments for further consideration.

d. Comment - Radiation Effects

Where are the data showing the effects of neutron radiation for the spectra, flux and temperatures which apply to the PRISM Cr-Mo containment vessel steel and its weld metal (refer to Section 5.2.1.3, page 5.2-5)? Refer to NUREG-0933, Generic Issue 15, "Radiation Effects on Reactor Vessel Support Structures," as an illustration of the effects of ex-vessel neutron radiation at low temperature on ferritic steel.

Will radiation have an effect on low-temperature embrittlement of austenitic stainless steel? Consider, for example, the ongoing research at the Argonne National Laboratory (Ref.: NUREG/CR-4744 Vol. 1, No.1, September, 1986). If radiation causes some metallurgical changes, will it have an effect on the corrosion-erosion behavior in the presence of liquid sodium? Has the effect of radiation on Inconel-718 been adequately considered since it contains cobalt (unless specially ordered) and boron (see pages 5.3-7, -21 and -21, where I-718 is mentioned)? Also, on p. 5.3-7, it is mentioned that Stellite 6 is being considered as a wear surface material when it is well-known (Ref.: E.I>Hunt, "Reducing Personnel Exposure by Reducing Cobalt Levels in Primary Systems," Nuclear Plant Journal, May-June, 1987, pp. 24-25 and 50) the resulting Co level is undesirable. Further discussion is warranted.

Response

The estimated fast fluence for the reactor vessel is 6.8XE12 n/cm which is well below the fluence levels which have been shown to affect the fracture toughness properties of ferritic steels. The containment vessel fluence levels will be 3 times less than the reactor vessel and should not influence the toughness properties of this vessel. The neutron fluence levels expected for the reactor vessel are considered too low to influence the toughness of the austenitic steel vessel also. The temperatures of the sodium are considered too low for corrosion/erosion effects to be important regarding any of the materials in the reactor system.

e. Comment - Pump Seal

Additional discussion regarding the reliability of the pump seal mentioned on page 5.5-21 is in order. What experience and/or laboratory data are available on the stated design?

Response

The shaft seal is a Byron-Jackson seal design based on the CRBR prototype pump test and development program. The design is also identical to the MONJU shaft seal which has been manufactured and tested.

**Comments for further consideration.

CASES OF INSUFFICIENT DETAIL

Comment f.1

One line carries steam to the turbine (see Figure 5.1-3, page 5.1-17, marked-up copy attached). The reliability of that pipe line must be high to avoid unscheduled shutdowns; not enough has been said in its defense.

Response

The main steam lines, including the single steam line, between the three steam drums and the power turbine are designed to high quality industrial grade requirements providing functional reliability consistent with the plants high availability goals.

Comment f.2

Regarding Item (1), Section 5.2.1.4, page 5.2-4, where will the readout be located and why should not the design include an independent check on the plug position?

Response

A number of instruments and administrative procedures are provided to determine the rotary position of the rotating plug. During the refueling operation, a triplicated set of position sensors provide position information to the PCS electronics located in instrument vaults adjacent to the Head Access Area. Local, diagnostic readout is available within either of these two PCS instrument vaults. A local readout is provided at a manual workstation within the refueling enclosure above grade during the refueling of the reactor module.

A machine engraved scale on the outer surface of the rotating plug will enable visual determination and/or confirmation of the position of the plug.

The zero, seated, locked position of the rotating plug will be sensed by the quad redundant (4 sensors, etc.) Reactor Protection System. The RPS will prevent any control rod drive attempt to pull control rods if the rotating plug is not in its zero, seated, locked position. Readout of the RPS information is available within each of the four RPS Instrument Vaults adjacent to the Head Access Area and at all locations served by the DHTS.

Written administrative procedures; including checksheets, visual inspection, and thorough training will back up the automated, redundant position sensing and display.

Comment f.3

On Figure 5.2-1, page 5.2-21 (called-out on page 5.2-5), add data under the words "REACTOR VESSEL" to indicate its dimensions as has been done for the containment vessel.

**Comments for further consideration.

Response f.3

The following dimensions will be added to Figure 5.2-1:
"18 ft - 10.0 O.D.
2.0 wall thk."

Comment f.4

On page 5.2-6, third paragraph from the top, the text implies that there may be some changes in the reactor vessel dimensions. If so, how might they change and for what purpose?

Response

The text did not intend to imply potential or possible changes to the reactor vessel dimensions. Its intent was to describe important factors that influenced the overall dimensions of the reactor vessel. Changes to Section 5.2.2.1 provide this clarification.

Comment f.5

Further discussion is needed on page 5.2-6, bottom paragraph, to describe the "special measures" which will be applied to the containment vessel surfaces.

Response

These special measures include coating the surfaces with oxides produced during required post-weld heat treatments in air. The vessel surfaces will be protected during installation to insure the coating is not damaged prior to plant startup.

Comment f.6

With regard to the bottom paragraph on page 5.2-8:

- Where is leakage detection discussed?
- What happens if the leakage exceeds 0.1% of the gas volume per day?
- How is the leakage gas (when less than the criterion) handled?

Response

The leakage value of 0.1% of the gas volume per day is the limit set for the closure including the many penetrations. This will be the technical specification limit that will be checked prior to the reactor returning to operation. Should the limit be exceeded, the startup procedure will be stopped until the leaks are detected and corrected.

**Comments for further consideration.

It should be noted that the last part of the paragraph should read as follows:

> All penetration joints will have welded seals of some type during normal operation. Because refueling operations require joint separation at the three aforementioned penetrations, reliable mechanical seals will be used for them, but provisions have been made in the design to make and break the welded seals.

No special provisions have been made to handle the leakage gas because the testing for gas leakage will take place following a refueling operation at which time the reactor cover gas will have been cleaned up prior to refueling. What gas may leak into the head access area (HAA) will be of no radiological consequence.

Comment f.7

Considering the problems LWRs have experienced as a result of choosing high-strength steels for bolting applications, the choice cited on page 5.2-9, second paragraph, demands some elaboration to illustrate how historical problems are to be avoided.

Response

The high temperature, high pressure steady loading conditions applying only to LWR bolts, whereas only low temperature Level D conditions loads exist for the PRISM closure bolting. Therefore, the noted historical problem does not apply.

Comment f.8

On page 5.2-12, bottom paragraph, the words "guard" vessel are used. It was assumed that that was synonymous with "containment vessel." If not, explain. In any case, when a different term or phrase is introduced for a component, as a courtesy to the readers, relate it to the foregoing.

Response

See change to Section 5.2.3.

Comment f.9

Consider the stress states to be imposed on the ferritic steel containment vessel and the temperatures it will see (e.g., from Tbl. 5.2-6, page 5.2-19, P=37.7 psia at 785°F). Is there cause for concern over biaxial creep? As part of Section 5.3.1.1, page 5.3.1.1, page 5.3-1 ff, there may be a need to include discussion of low-temperature embrittlement and erosion-corrosion effects for the stainless steel components and the influence of radiation on those phenomenon. These issues have been mentioned separately in earlier items. Is there potentially some synergism?

**Comments for further consideration.

Response f.9

During normal operation, the containment vessel operates at temperatures below 700°F and is designed according to the ASME Code Section III, Subsection NB. During duty cycle events associated with loss of the normal heat sink where the heat loss is only by RVACS, portion of the containment vessel will be exposed to temperatures higher than 700°F. The design for these events follow the ASME Elevated Temperature Code Case N-47 which accounts for the creep effects through time-dependent stress limits (S_t) in addition to the time-independent stress limits (S_m). The duration of the RVACS events, however, amounts to about 1000 hours for which the creep damage is negligible and the containment design is still governed by time-independent primary stress limit (S_m) rather than the time-dependent stress limit (S_t).

The ex-core reactor structures including the reactor vessel and the containment vessel are shielded from the core environment sufficiently to limit irradiation embrittlement and retain the fracture toughness and ductility necessary for the application of the ASME Code.

Comment f.10

On page 5.3-5, bottom paragraph, the words "austenitic steel" are used; should it read: "austenitic stainless steel"? Locking pin, bolts, studs, nuts, etc., by welding, although a common practice, can be a problem from more than one point of view. Applied to some materials, cracking can result; when ASME code welding qualification is required, the size of the weld may prohibit meeting code requirements. How have such considerations been taken into account with respect to the method called out on page 5.3-7, paragraph three?

Response

On page 5.3-5, last paragraph, the word stainless is added as "austenitic stainless steel" in Section 5.3-2. Consideration for implementing the method of lock welding assembly pins for the core former ring will be applied as the design matures and details of specific parts are identified.

Comment f.11

Regarding the B_4C leakage mentioned on page 5.3-10, what is the consequence of such leakage?

Response

Other than slight local diminution of shielding, no deterimental consequence of B_4C from the fixed shielding can be envisioned. The second barrier cannisters take the tubes out of the main sodium flow streams so that leaking boron carbide should just sink to the bottom of the cannisters. Should the B_4C find its way out of a cannister, it should again sink to the bottom of the reactor vessel. Any very fine particulate which might be carried in the flow stream would plate out on a boundary surface.

**Comments for further consideration.

Comment f.12

What is the material of the vessel liner (page 5.3-12, second paragraph) and what criteria were employed in choosing it?

Response

The reactor vessel liner material is type 304 stainless steel. It was chosen for compatibility with other reactor internal structures and for its capability to withstand the environment of its location within the reactor vessel. Its design and construction requirements and criteria will be as stated in Section 5.3.2.5.

Comment f.13

On page 5.3-21, the materials cited as going into the shroud tube assembly include Type 316 stainless steel and Inconel-718 which have (approximately) linear coefficients of thermal expansivity of 7.2 per °F, and 9.1 per °F, respectively. Show that the assembly can tolerate the differential thermal expansivity or rethink the material choices (e.g., will the expansion of the I-718 induce radial creep of the stainless steel and some resulting malfunction of the assembly?).

Response

The differential thermal expansion of Inconel 718 and Type 316 SS was recognized and accounted for in the UIS/component design including the shroud tube design. The design is being modified to improve the seismic performance and thermal response times. However, the design changes will not affect the adequacy of the Inconel 718-SS316 interface. The operating temperatures are sufficiently low to preclude significant creep in SS316.

Comment f.14

Why are Type 304 ASME stress limits used in Table 5.3-3, page 5.3-36 (call-out on page 5.3-30) when the chosen material is Type 316 stainless steel?

Response

Type 304 ASME stress limits were used in the initial reactor vessel design evaluation to allow for its possible selection as an alternate to SS316. The stress limits have been updated in the current evaluations to conform with the present selection of SS316 as the reactor vessel material. This has no effect on the conclusions regarding the structural adequacy of the design since SS316 stress limits tend to be higher than those for SS304.

**Comments for further consideration.

Comment f.15

Regarding the pump discussed on page 5.4-8, if it locks ("freezes") will it over-heat? If so, by how much and what safeguards are in place to accommodate such an event?

Response

The PRISM reactor has four EM pumps. An EM pump has no moving parts and is not subject to freezing (lock up) associated with mechanical pumps.

Comment f.16

There seems to be no detailed description of the materials of construction of the IHX (Section 5.4.2.3, page 5.4-11), please provide such a description with particular attention to fabrication methods (welding, forming, heat treating, etc.)? Also, in the same section, will the stainless steels to be used in the IHX be in compliance with NUREG-0313, Revision 2?

Response

The IHX is constructed of Type 304 stainless steel throughout, except it has Inconel 718 piston ring seals attached to the bottom nozzles for fit with the seal plate within the reactor vessel.

Construction of the IHX will be in accordance with Subsection NB of Section III of the ASME Code and the 304 SS material will meet the requirements of Regulatory Guide 1.44. Fabrication of the IHX has been studied for sequencing and feasibility. Descriptions of fabrication methods and procedures will be provided with detail designs.

Comment f.17

In Section 5.5.1.2.4, page 5.5-8 ff, in Item (1) the text lists the materials of construction for the IHTS piping then Item (3) cites a different material. Why is the tank not to be fabricated from 2-1/4Cr-1Mo steel?

Response

The IHTS system piping, pump and expansion tank will be constructed of 304SS. The more costly, higher strength 316SS may be used in high temperature regions if it is necessary or advantageous to do so. The 304SS IHTS piping will connect to the steam generator shell which is 2-1/4Cr-1Mo ferritic steel. The 304SS IHTS piping will also connect to the sodium dump tank which is SA-533, low alloy ferritic steel. The steam generator material, 2-1/4Cr-1Mo, was chosen based on the steam generator water service and the sodium dump tank material, SA533, was chosen because of its impact properties. The metallurgical transition between

**Comments for further consideration.

Response f.17 (Continued)

the austenitic stainless steel and the 2-1/4Cr-1Mo or low alloy ferritic steel is made utilizing a trimetallic spool piece. The intermediate coupling material is Incoloy 800H (Ni-Fe-Cr) nickel alloy. Inconel 82 is used as the weld filler material between the ferritic steel and alloy 800H and 16-8-2 stainless steel filler is used betwen Incoloy 800H and the austenitic steel.

Comment f.18

Why is there an apparent uncertainty about the material to be used in building the IHTS expansion tank (Section 5.5.2.4, page 5.5-22); the text shows "304/316 stainless steel."

Response

The IHTS expansion tank will be fabricated basically from 304SS. The designer may use the more costly, higher strength 316SS in high temperature regions of the pressure boundary (such as at the SG vent nozzle) if it is necessary or advantageous to do so.

5.30** Comment

Miscellaneous

a. Refer to page 5.1-11; if there has been a S/G failure and a resultant Na/H_2O reaction, will the flashed steam vented to the atmosphere carry radioactivity? If so, discuss.

b. For the change in hydrogen concentration levels noted on page 5.1-12, at what level of detection will the reactor be shutdown, by what controls/actions for what reasons and with what results?

Response

a. In the event of a sodium-water reaction, the pressure in the SGS is reduced from 1000 to 200 psig by isolating the system and opening the water dump valves and safety relief valves. When the system pressure drops to 250 psig the power relief valves are closed and at 200 psig the water dump valves are closed. Inert nitrogen gas is injected into the steam system at 200 psig to prevent sodium back flow through the tube leak, to inert the SGS and to purge the SG tubes. Nitrogen gas is also injected into the sodium dump system at 50 psig to purge the system of hydrogen and prevent an explosive mixture of hydrogen and air from forming. Throughout the steam/water dump there is no sodium flow into the SGS so only steam is vented. In addition, the intermediate sodium is not radioactive, therefore, no radioactivity is vented to the atmosphere.

* Comments requiring textual improvement.
**Comments for further consideration.

5.30** **Response (Continued)**

b. The IHTS protection against a sodium-water reaction is the dual rupture discs located on the SG shell. The hydrogen leak detection system is used only to allow the reactor operator to monitor the system for small leaks. No automatic action is taken in the event of a leak; the plant operator takes any necessary corrective action.

These hydrogen leak detectors are located on each of the sodium streams from the SG. Present state-of-the-art leak detectors do not always provided a reliable signal. Two leak signals from the three detectors on a given line are required to provide a "confirmed leak."

Upon receipt of a confirmed leak signal the operator determines the leak size from the rate hydrogen concentration increases in the IHTS.

For low level leaks (less than $2x10^{-5}$ 16/sec), the operator will reduce power to 25%, continue to monitor the leak and prepare for a planned reactor shutdown. Low level leaks have a definite tendency to self-plug. It is desirable to have the leak remain open during plant shutdown to aid in leak location and repair.

For intermediate level leaks ($2x10^{-5}$ to $6.5x10^{-3}$ lb/sec), the operator will initiate a controlled reactor shutdown followed by SGS isolation and blowdown. Leaks of the intermediate size will usually remain open after reactor shutdown.

For high level leaks (greater than $6.5x10^{-3}$ lb/sec) the operator must respond to prevent severe leak damage and possible escalation into a large leak capable of bursting the rupture disc. The operator will initiate a reactor scram followed by SGS isolation and blowdown. The IHTS sodium will then be drained and steam generator repaired.

5.31* **Comment**

Will the ability to withstand the SSE be demonstrated by vibrating the reactor prototype or evaluated in some R&D project?

Response

See Section 5.1.1.

5.32* **Comment**

The argon gas between the containment and reactor vessel is held in place by the compression load on the seal between the flange and the closure.

- Would the plant be shut down in the event of an argon leak?

* Comments requiring textual improvement.

5.32* <u>Response</u> **(Continued)**

See Section 5.2.2.2.

5.33* <u>Comment</u>

It is stated that the rotatable plug is in a bolted down position during normal operation. Are these bolts locking the plug into the containment during normal operation? Current figures in the PSID do not show the locations of these bolts.

5.33* <u>Response</u>

See Section 5.2.2.3.2.

5.34* <u>Comment</u>

How would B_4C leakage from the fixed shields be cleaned up? How would the B_4C be removed from the system and the shields replaced?

<u>Response</u>

See Section 5.3.2.3.

5.35* <u>Comment</u>

Will leakers be placed in the storage rack or be removed before they have decayed enough to permit dry handling?

<u>Response</u>

See Response to Comment 4.3.

5.36* <u>Comment</u>

Where are the incore flux monitors?

<u>Response</u>

See Section 5.3.2.11.

5.37* <u>Comment</u>

Could an SSE bend the latch on the control rods enough to bind up and stop the insertion process?

* Comments requiring textual improvement.

5.37* Response

See Section 4.5.1.

5.38* Comment

Operational requirements on these plants must be between 40% and 100% power because of the assembly wobble that occurs until the core restraint ring contracts the TLPS. Hence, the requirement of 25% to 100% power range for the plant as stated in the PSID is *not* met. Does this statement in the PSID need to be changed?

Response

See Section 4.2.2.3.

5.39* Comment

Have the COMMIX predictions of a well-mixed hot cold plenum been verified by your scale model plant test at ANL?

Response

See Section 5.4.3.1.3.

5.40* Comment

Is fresh sodium tested for purity before being pumped into the IHTS or the PHTS?

Response

See Section 9.5.2.3.3.

5.41* Comment

How is the dimensional gauging performed when inspecting components in the reactor vessel?

Response

See Section 5.8.1.2.

5.42* Comment

Please supply a drawing of the seismic isolators and indicate how they support the reactor module. Where are the modular support brackets:

Response

See Section 5.1.1.

* Comments requiring textual improvement.

6

6.1 Comment

Given the number of penetrations into the Head Access Area (HAA), and the number of seals necessary to form the containment boundary, has GE considered using a secondary containment around the HAA?

Response

Consideration has been given to a secondary containment for the closure penetrations. However, the designed leak protection for the closure and penetration (i.e., as discussed in the responses to Comments 5.5 and 5.29**f.6) provides significant margin to satisfy the containment requirements of Sections 1.2.1.2.25 and 1.2.1.2.30 and the evaluations of Sections 3.1.5.1 and 3.1.5.7. Considerations of secondary containment is continuing as the design matures to assure that the margins are retained.

6.2 Comment

On page 6.2-8, Paragraph 2 begins, "Provision is made for isolation of the HAA by means of electrically-operated dampers in HVAC ducts." How is the isolation actuated and how is the HAA cooled under these conditions?

Response

The dampers are actuated by temperature, smoke, radiation or sodium aerosol sensors at appropriate locations. Each HAA is cooled with unit coolers connected to central dump heat exchangers/compressors for each power block. Freon is the working fluid in the coolers.

6.3 Comment

Radiological assessment results were obtained for source terms based on the following release fractions to the cover gas following 100% core melt:

Noble Gases:	100%
Halogens (Iodine and Bromine):	0.1%
Particulates (Cesium and Rubidium):	0.1%
Transuranics (Plutonium):	0.01%

Please supply or identify reports to support the assumed sodium retention of fission products.

Response

The cover gas source terms used in the radiological assessment were estimated for an oxide core and have not been updated for the metal core design. However, because of the similarity in calculation

(6.3 Response - continued)

assumptions and methods, the release fractions quoted are not expected to change as a result of updated analyses including considerations given to 100% core melt.

The release fractions are dependent upon (i) the liquid-to-gas transport processes appropriate to the fission product species being considered, (ii) the in-vessel conditions especially the temperature of the upper sodium plenum, and (iii) the volumes of cover gas and primary sodium. The absolute source term is also dependent upon the fission product inventories. There are a range of transport processes to be considered some of which can be derived from basic thermodynamic principles, e.g., Weast (Reference 1); others are based on experimental data such as Castleman, et al. (Reference 2), Jordan and Ozawa (Reference 3), Schultz (Reference 4) and Berlin, et al. (Reference 5). Where alternative approaches are considered for a given species the most conservative method is within the release fraction quoted. The updated metal core will be re-evaluated in order to confirm that its release fractions are indeed enveloped by the quoted values. The range of methods will be coordinated into a single reference and include a comparison where alternatives exist (Reference 6).

References

1. R.C.Weast, ed., Handbook of Chemistry and Physics, 55th Ed., CRC Press, Cleveland, Ohio (1974).

2. A.W.Castleman, Jr., I.N.Lang, and R.A.Mackay, "Fission Product Behavior in Sodium Systems," in Akali Metal Coolant, Proc. Symp. on Alkali Metal Coolants - Corrosion Studies and System Operating Experience, Vienna, November 28 - December 2, 1966, IAEA, Vienna (1967).

3. S.Jordan and Y.Ozawa, "Fuel Particle and Fission Product Release from LMFBR-Core Catcher, "CONF-761001, Vol. IV, p. 1924 (1976).

4. W.Schultz, "Fuel and Fission Product Release and Transport from Hot Sodium Pools," Proc. of the International Mtg. on Fast Reactor Technology, August 19-23, 1979, Seattle, WA Vol. 3.

5. M.Berlin, et al., "Evaluation of the Sodium Retention Factors for Fission Products and Fuel," Proc. of the LMFBR Safety Topical Meeting, July 19-23, 1982, Lyon, Ecully, France, Vol. III.

6. R.W.Tilbrook, report on LMF source term methods, ANL, to be published 1988.

6.4* Comment

Appropriate sections in Chapter 6 should indicate the seal welded areas.

Response

See changes made to Sections 6.2.1.2, 6.2.1.3, and 6.2.2.

6.5** Comment

The staff notes that Inservice Inspection (ISI) is not well defined for complicated goemetries such as the closure head. Inquiry of the ASME Section XI Committee should be initiated to clarify this situation and prevent future difficulties.

Response

The closure head for the PRISM reactor is a simple design and can be readily inspected to ASME Section XI requirements. The inservice inspection needs for the closure head are well understood and no further clarifications are needed.

The closure head is comprised of a twelve-inch thick, solid stainless steel plate. The closure head has large openings for the installation and possible removal of the two IHX's and four EM pumps. These components have flanged covers and underside shielding which close the opening. All covers are bolted down and the joints are enclosed in a welded cover.

The rotatable plug (RP) is a non-integral part of the reactor closure for which special connections and seals are provided. Its structure is essentially the same as for the rest of the closure head, but at its circumference it has an upstanding rim and flange which is concentric with a similar rim and flange around the center opening in the stationary part of the closure. During power operation the rotatable plug is sealed to the closure head by a seal welded cover.

Inservice inspection of the PRISM (ISI) closure head is based on the current requirements of the ASME Section XI, Division 3 Code. This ISI section of the ASME code was developed around the CRBRP design to which PRISM is similar. The specific area of the ASME code that covers the PRISM closure head is Category B-C, Cover Gas Retaining Components, Item B3.11, Structural and Seal Welds (in austenitic stainless steel only). The method of examination for this component of the design is continuous monitoring.

* Comments for textual improvement.
**Comments for future consideration.

7

RESPONSES TO NRC COMMENTS ON PRISM PSID CHAPTER 7

7.1 Comment

Design of Man-Machine Interface

Describe the design process used to identify the man-machine interfaces needed to operate the plant for normal operation, transients, and accidents. Of specific concern to the staff are human factors issues as they relate to General Design Criterion (GDC) 19, Control Room (10 CFR 50, App. A). Provide the rationale for any exceptions or deviations from the requirements in GDC 19. Identify and discuss the number of operators and their duties for each man-machine interfaces. Also, identify all work stations from which the operator may manually initiate a trip of a reactor module.

Response

The basic man-machine interface design is derived from GDC13, GDC19 and IEEE 603. A review of these documents have identified three basic machine interface areas:

- Central Control Room (CCR)
- Auxiliary Shutdown Console in Reactor Service Building (RSB)
- Reactor Instrument Vaults (Local)

The central control room (CCR) operator performs plant mission management tasks. His role is primarily a supervisory one where he monitors and confirms plant behavior under normal and abnormal conditions. He can change the controller setpoints and thereby control the power plant, and he is responsible for releasing hold points for semi-automatic plant operations. He can execute discretionary manual actions if the automatic controls are not working properly. He can also, by manual initiation, bring the reactor to cold shutdown by requesting an RPS scram or a PCS fast runback. The scram request is not safety-related. Any plant transient that challenges plant safety will cause safety protection trip parameters to be exceeded and the RPS will automatically scram the reactor. The central control room is also normally the center from which communications with the roving operators and off-site locations is maintained.

The auxiliary shutdown console in the Reactor Service Building (RSB) is an alternate area from where the operator can achieve and maintain plant shutdown conditions in the event the control room becomes uninhabitable or the control room equipment becomes inoperable. The controlled shutdown of any reactor can be initiated by a manual scram request or fast runback from this console just as from the main control room. The RSB facility is designed to protect the operator from the earthquake, fire, smoke and other noxious airborne contamination; however, the control electronics in the auxiliary shutdown console, as in the control room, are not safety-related.

The reactor instrument vaults contain the RPS electronics for initiating reactor scram. The RPS electronics for each reactor has four divisions, and each division is physically separated and located in its own room within the reactor vaults. The electronics are

7.1 Response (Continued)

safety-related and are qualified Class 1E for this function. Provision is also made for a safety-related man-machine interface in the RPS electronic panels. Access to safety-related equipment and functions including manual reactor scram can be made from this MMI. A local display is provided for monitoring.

All three man-machine interface areas are being designed to assure that operators do not receive radiation exposure during accident conditions in excess of the limits specified in GDC 19.

PRISM design is presently at a conceptual stage and the exact number of operators required and their precise duties have yet to be determined. The current concept provides one operator for each PRISM Power Block (or Nuclear Power Unit) acting in a supervisorial (cognitive) mode over a highly reliable and intelligent Plant Control System, and a number of roving operators that provide support functions from the RSB and the RPS instrument vaults.

In the detailed design phase, task analysis will be performed that will meet the intent of NUREG 0700 and NUREG 0800 (SRP 18). The outcome of the study will provide the detailed man/machine allocations, and will quantify the number of operators and their tasks.

7.2 Comment

Accident Monitoring

Identify the process functions, process variables, and plant systems (consistent with the preliminary design) that operators must monitor to meet the intent of GDC 13, Instrumentation and control (10 CFR 50, App. A), and Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plant to Assess Plant and Environs Conditions During and Following an Accident.

Response

The following is a list of variables used for PRISM post-accident monitoring. The variables are consistent with an application of R.G. 1.97 to a liquid metal reactor such as PRISM. The monitoring of these variables is in agreement with GDC-13.

PRISM ACCIDENT MONITORING VARIABLES

Variable	Range	Purpose
Reactivity Control		
Neutron Flux	10exp-6% to 100% full power	Detection, Trip
Control Rod Pos.	Full in or not full in	Verification
Core Cooling		
Cold pool temp.	0 - 2000°F	Detection, Trip
Core outlet	0 - 2000°F	Detection, Trip
Coolant Level	0 - 20 ft	Detection, Trip
Core inlet pres.	0 - 200 psig	Detection, Trip
RVACS exit temp.	0 - 500°F	Monitoring
RVACS air mass flow	0 - 80 lbm/sec	Monitoring
Reactor Vessel Integrity		
Sodium Leakage	Yes/No (Spark Plug)	Monitoring
Cover Gas Pressure	10exp-5 - 20 psig	Monitoring
HAA Radiation	1 R/hr - 10exp7 R/hr	Surveillance
Environs Radiation	10 exp-3 R/hr-10exp 4 R/hr	Surveillance
Head Penetration Valves	Open/Closed	Verification
Rotating Plug	Open/Seated	Verification

7.3 Comment

Reactor Control System Design

Identify and discuss the design process that develops the reactor control system. Of specific concern to the staff is the stability margin of the control system. Discuss the design of the stability margin of the control system as a function of shim rate, power level of operation, and the positive reactivity feedback due to sodium.

Response

Design of the Reactor Control System (RCS) first involves an analysis of control system requirements, and iterative refinement of a prototype control system. Iterative modeling and simulation of control strategies and hardware/ software modeling results in definition of a prototype control system. Prototypes are procured and hybrid tested using a plant simulator. The RCS is then validated in a plant demonstration test.

Plant Control System design to date has concentrated on the overall multi-module design aspects of PRISM. Only Proportional-Integral Derivative (PID) controls have been utilized to date in simulations of the Reactor Control System. An inner flux controller loop with an outer reactor core outlet sodium temperature controller has been used.

7.3 Respose (Continued)

Detailed sensitivity studies of the PRISM Reactor Control System will be made when the state variable RCS controller design is evaluated. Results of 10% step temperature changes to the PRISM RCS at 90% initial reactor module power are available. They show that for the controller gains used, the controller has at least a 20 db gain margin, is stable, and the reactor outlet temperature controller response:

1) improves with increasing rod speed
2) improves with increased positive sodium density feedback (within limits, of course)
3) improves with increasing power level (observed from step and ramp response studies of multi-module control)

The PRISM RCS step response results are listed in the following table.

CASE	0 To 63.2% Of Final	10% to 90% Of Final	Time To Peak	Percent Overshot
Base Case (15 in/min rod speed)	9.6	9.4	17	22
Base Case Except (9 in/min rod speed)	14	14.6	~28	36
Base Case Except (Na reactivity coefx2)	9	8.9	15	26
Base Case Except (Na Reactivity coef=0)	10	10	~19	26
Base Case Except (Temp. controller gain x 10, or 20 DB)	9.6 (Oscillating but still stable, shows a >20 DB gain margin exists)	9.4	~30	~132

Future reactor control system design will incorporate signal validation and analytical redundancy to assure the reactor module state is correctly estimated. Multivariable control (including the feedback of un-measurable variables such as reactivity) using either non-linear or piecewise linear dynamic models of the physical process and controllers will be utilized to improve the reactor control system stability margins.

The results of sensitivity studies for the CRBRP are useful for backup estimates of rough order of magnitude effects and trends. The CRBRP sensitivity studies showed the reactor temperature controller was relatively insensitive to nominal changes in control rod worth (and therefore rod rate) and Doppler coefficient (and therefore sodium density coefficient). For fixed controller gains and compensations, a considerable decrease in stability margin with decreasing power level was predicted.

7.3 Response (Continued)

Stability Margin as a Function of Power Level

A CRBRP PID controller simulation with an inner flux loop and an outer reactor core exit temperature control loop predicted the following gain and phase margins as power level was varied.

	Flux Loop		Temperature Loop	
Power Level	Gain Margin	Phase Margin	Gain Margin	Phase Margin
100%	∞db	128°	27.2 db	95°
40%	∞db	127°	18.6 db	80°
20%	∞db	125°	11.2 db	50°

Stability Margin as a Function of Shim Rate

An increase in control rod speed increases the inner flux loop gain. This decreases the phase lag in the temperature loop and provides additional phase margin. CRBRP sensitivity studies showed this effect in an alternate way. It was shown that temperature loop gain and phase margins increased at all power levels with increased control rod worth. A doubling of rod worth increased the gain margin by 5 db and phase margin by 2° at 100% power. These would be an equivalent effect for a doubling of rod speed.

Stability Margin as a Function of Positive Sodium Temperature Reactivity Coefficient

The positive sodium density reactivity feedback relative to zero core power is shown in the Table attached to the response to comment 4.28 on page F4-52. It is seen that the major positive reactivity effect is due to the sodium density reactivity feedback. The positive sodium density reactivity is no more than one-third of the equally rapid negative reactivity contributions from Doppler and fuel axial expansion. CRBRP studies showed that a fourfold increase in the Doppler feedback coefficient decreased the temperature controller gain margin by only 4 db, and did not change the phase margin. From this it can be inferred that a small increase of positive sodium density reactivity would increase the temperature controller gain margin.

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7.4 Comment

Operator Workloads

One reactor operator monitors a power block. Each power block contains three reactor modules. Discuss and identify operator workloads as a function of monitoring functions, manual controls, and emergency response procedures.

Response

In summary, the operator has no nuclear safety-related function to perform in relation to a PRISM reactor module. Each PRISM Reactor Protection System is designed to function automatically and independently from any other system - including the reactor/power block operator. Each PRISM RPS is backed by the inherency designed into each reactor module. All nuclear safety-related events that challenge the nuclear safety of a PRISM reactor module are acted upon by the PRISM RPS.

Each Reactor Protection System is concerned only with the nuclear safety of the reactor module it is protecting. In response to a nuclear safety parameter challenge, the RPS will take all actions necessary to shut the reactor down. There are no further actions that can be taken from the Control Room or the Remote Shutdown Facility with respect to the primary reactor system, to shut the reactor down. There are no emergency or auxiliary core cooling systems to be initiated or monitored for the PRISM plant. No action in the Balance of Plant is required for the nuclear safety shutdown of a PRISM reactor module. The operator and the Plant Control System are responsible for any actions required to protect the investment in the Balance of Plant and for communications with the grid controller for any reallocation of load or power reduction.

If a reactor module trips and the total plant is running below its maximum generating capacity, the PCS control strategist will adopt an appropriate load redistribution and if possible retain the requested plant power generation requested demand. The PCS will however automatically acknowledge the reactor module in a scram state and reduce the subsystem controller setpoints on all control subsystems requiring adjustment within that power block.

The PRISM plant control system design is still in the conceptual design stage. At this point in time in the design it is not possible to do a detailed operator work load analysis. However, a gross task workload evaluation has been performed in FY87 as part of a PRISM Plant Automation Assessment Study. The study will be issued in September 1987. As part of this study an assessment was made of the operators workload for two typical events that challenge the power block operator with a high workload, and it was determined that with the planned level of automation, one operator could indeed control a power block.

It was also determined from this assessment that the operator performed the following category of actions:

7.4 Response (Continued)

- Observing and monitoring the plant under normal and abnormal conditions. Alerting appropriate personnel as required.
- Granting permissives for start of automated operational sequences, and releasing of hold points.
- Manual control (problem resolution) in the event the automatic control system encounters an unknown state.
- Discretionary readjustment of controller setpoints depending on plant state.
- Communicating with shift supervisor, the roving operators, the maintenance staff and other power block operators.

7.5 Comment

Local Control Stations

Describe the control law used by local control stations upon loss of communications with the plant control system. Furthermore, describe how local control stations utilize modern control theory and mathematical models of the plant.

Response

Since local control stations will communicate with the power block and plant level controllers via a redundant bus, the probability of loss of communications will be very low. The loss of communication will be detected by both the local controller and the higher level controllers through loss of the "handshake" data exchange between the controllers. Since the probability of this type of event is very low, fast runback and shutdown of the affected module could be utilized without significantly affecting the plant availability. However, since the isolated local controller still will retain its investment protection function, continued operation at the last setpoint is currently judged to be acceptable (except when the module was undergoing fast runback or module trip, which would be carried to completion).

The local control stations will contain models of the respective portion of the physical plant being controlled. These models are useful as both 1) an "estimator" or "observer" of the local physical systems operation (calculating key parameters such as reactivity, which cannot be measured directly; monitoring plant changes with time compared to the mathematical model) and 2) to provide additional non-measurable feedback variables which can be used to improve the local controller performance. The feedback gains can be selected to meet desired performance specifications.

The local controllers are coordinated by a power block optimal controller which takes into consideration interactions between the local controllers. By dividing the power block into local

7.5 Response (Continued)

controllers, matrices used are reduced, thus providing a large reduction in computational requirements.

Details to utilizing modern control theory and mathematical models for control of local plant systems is described in R. A. Kisner and G. V. S. Raju "Automating Large-Scale Power Plant System: A Perspective and Philosophy," ORNL/TM-9500, dated December 1984.

7.6 Comment

Generic Control Engine Concept and The Plant Control And Protection System Concept

From our initial review, it appears that the control engine implements algorithms to automate process control. The control engine also automates the plant's response to failures in plant systems and components. If these goals are successfully achieved, the control engine should minimize challenges to the plant's safety systems.

The scope of the control engine covers many functions. Identify and discuss the role of all humans required to interface with the control engine. Furthermore, map the functions identified in Figure 7.7-5, Conceptual Control Engine Model, into the hardware and functions (control center, technical support center, information and management center, remote shutdown, operations support center, administrative and maintenance offices) identified in Figure 7.7-1, PRISM Plant Control and Protection System. In addition:

a. identify and discuss the role of the control engine in accidents, if any;

b. describe how the control engine initiates a trip request to the reactor protection system;

c. identify and discuss control strategy validation within the control engine for safety functions, if any;

d. identify and discuss the scope of maintenance planning within the control engine; e.g., does it cover Class 1E equipment and instruments?

e. identify and discuss how the control engine uses technical specifications of operation, if applicable;

f. describe the design, development, and test program for the control engine as well as the design verification and validation program; and

g. describe the role of the operator, the scope of the control, decision, and diagnostic aid provided by the control engine. Also, identify the design limits of the control engine and how potential conflicts from the control engine are to be resolved by the operator.

7.6 Response

The role of the shift supervisor and main control room operators is to monitor for safe and economic plant operation, making decisions which appropriately direct the plant control (see item g.). At the remote shutdown facility, the operators role is to assure safe reactor module shutdown and to monitor the plant. The role of the local, roving operators is to oversee local control functions, as required, under the direction of the control center operators. The role of the technical support center personnel is to advise management and operations on the status of the plant and to provide corrective action recommendations. The role of the operations support center personnel is to obtain necessary technical data and perform required maintenance during emergency conditions. The role of the emergency operations facility personnel is to provide responsibility for management of emergencies, and to coordinate and direct all offsite licensee activities.

An illustration of the relationship of the control engine functional interfaces with typical plant facilities is shown in Figure F7.6-1.

a) The role of the control system is to assure correct and economic operation of the plant, and following detection of undesirable perturbations or plant states, to direct the plant to the nearest safe state. Accidents, in general, involve mechanical failures which prevent correct and economic operation at the original plant state. The plant control system ascertains the nearest safe plant state (generally a shutdown reactor module with tripped sodium pumps at the affected module level) and directs the sequence of operation to attain that state. The control system informs the plant operators of these automatic control actions and of the attainment of a new safe plant state.

b) The plant control system conceptual design has been recently modified to include a PCS directed fast module runback. Plant control system requests for reactor protection system directed reactor module trip have been deleted.

c) All safety functions are performed by the reactor protection system. The RPS trip logic is simple and direct and once initiated is carried to completion without validation.

d) The maintenance planner will be added in phases, as determined by the user utilities requirements. The scope of the maintenance planner in providing a dynamic schedule of the maintenance programs is to 1) plan a preventative maintenance schedule, 2) incorporate corrective maintenance into the overall maintenance plan, and 3) issue the maintenance plan. Planning is included for both mechanical plant process components and control and safety system components. It will include Class 1E equipment and instrumentation.

e) Technical specifications will be included in the knowledge base of the control system. Technical specifications and other limits will be used by the plant performance analyzer function. Here

7.6 <u>Response</u> (Continued)

the correctness and economy of operation is examined in detail. The proximity and rate of approach to limits will be calculated.

The limits are also included in an abbreviated form in the control validator function. Here predicted performance is assessed, and potential strategies are rejected if key limits are estimated to be exceeded or approached at an unsatisfactorily high rate.

The Decision Support and Control Strategist functions also utilize technical specifications in their assessment and selection of strategies.

f) The design, development and test program for the control system will require the following steps:

- o Develop conceptual design
- o Develop plant simulator
- o Test design using simulator
- o Develop application software
- o Specify, procure prototype hardware (H/W) and operating system (S/W)
- o Load in application S/W
- o Interface prototype with simulator and retest
- o Optimize design, verify and validate design with simulator
- o Procure final H/W and S/W for prototype safety/power demo test
- o Perform prototype safety/power demo test and verify controller performance

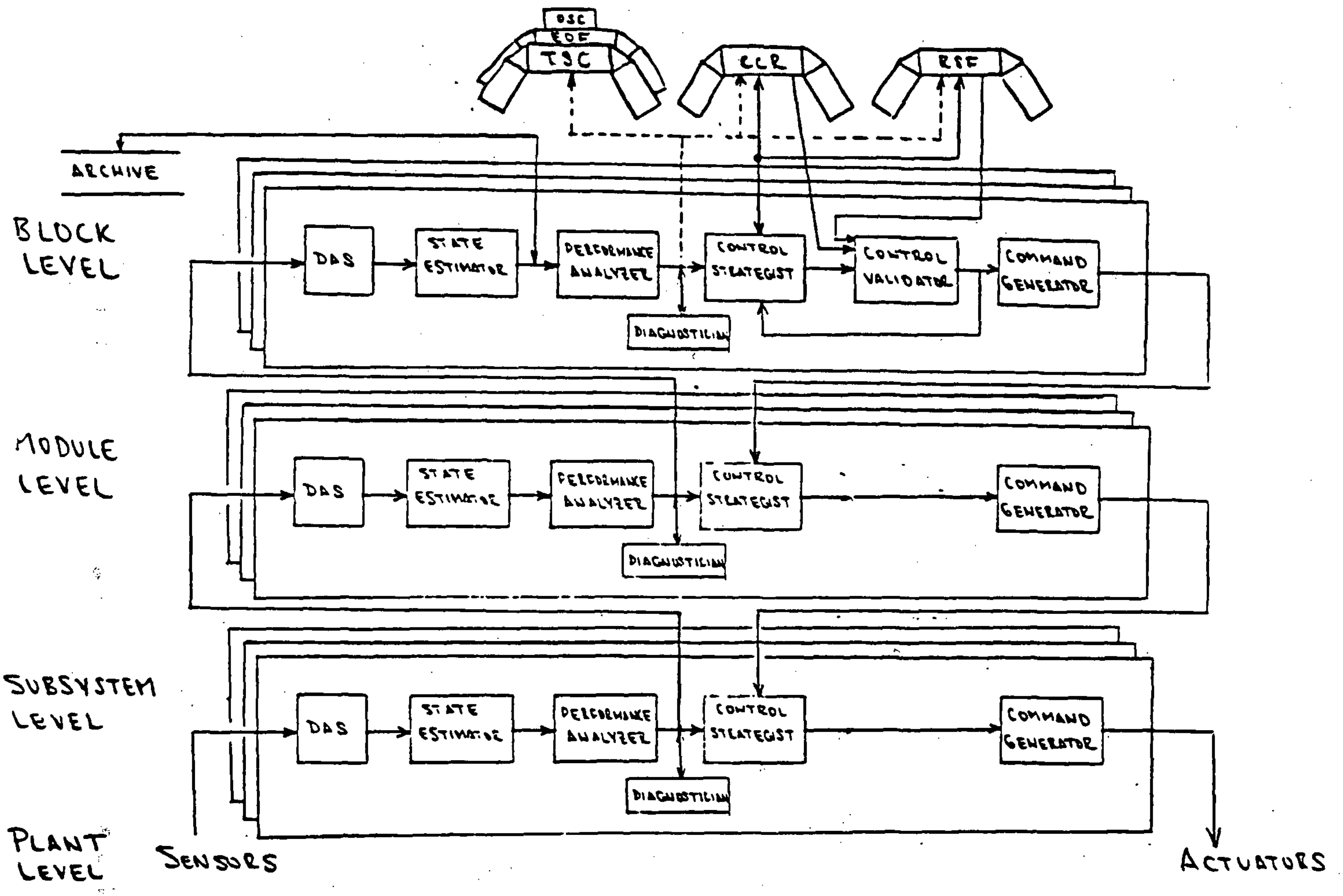
g) The operators role is that of a decision maker and permissive granter. The human function is "minds-on" rather than "hands-on." The most important role is to take care of the unforseen, to execute complex reasoning with less than complete information, to exercise judgment, and to troubleshoot and bypass possible equipment malfunctions.

The scope of the control, decision, and diagnostic aid (Decision Support) function of the control engine includes 1) filter and process data into an informational form useable by the operator, 2) provide access to the dynamic monitoring of the Performance Analyzer and Diagnostician functions, 3) provide displays of static information from off-line databases, such as operational manuals and instructional materials, 4) provide access to the Configuration Manager and Maintenance Manager functions to aid in planning of system reconfiguration for control or maintenance purposes, and 5) provide access to the Control Validator to test and validate operator proposed strategies.

Automatic control actions are only taken between known states. If an unknown state is encountered (due to an unforseen or unplanned for transient), the control system maneuvers the plant

7.6 Reponse (Continued)

to the next lower safe state and transfers control to the operator. The operator then performs the required control actions manually.



CONTROL ENGINE MAP

Figure F7.6-1

7.7 Comment

There are notable similarities between the data acquisition, performance analyzer, diagnostician and control strategist systems proposed in Chapter 7 and systems developed by the Halden Reactor Project in Norway. Discuss the areas of similarity, paying particular attention to experience gained through installation of the systems at operating power plants such as Ringhals in Sweden.

Response

GE and Halden Reactor Project have had a development program in the area of advanced control room design for the last 3 years, and many of the control engine concepts are being implemented in the Halden ISACS (Integrated Surveillance and Control System). The major developments at Halden have been in the area of advanced operator information and have not yet been extended to cover automatic feedback control. The Halden SCORPIO system (part of the ISACS) which is being implemented at Ringhals and Duke Power is a core surveillance system whose function is simulation of core power distribution in real time, comparison of efficient control strategies and evaluation of these through faster-than-real-time simulation. In this respect it is similar to the PRISM control engine's state estimator, performance analyzer, strategy generator and strategy validator functions. Both systems start with an estimation of the state of the plant based on a simulation of the process and fed with real time plant data. This function also yields many calculated variables useful in determining the reactor state. Deviations from the expected state and from the plant limits are then computed to determine anomalies (which are sent to the diagnostician for further evaluation) and to also prepare for control strategy selection. The plant state and diagnostics (decision aids) are displayed to the operator. The strategy generator, which contains various control strategy options, then chooses the optimum control strategy based on the plant state and plant goals that are either programmed in it or received from a higher level controller. For SCORPIO the goals are set by the operator and he also uses this input capability for what-if questions. The validator applies the strategy of a faster-than-real-time simulation of the process to predict the effect of the strategy and thereby validate it. The plant state, diagnostics (and operator aids), selected control strategies, and predictions are all sent to the operator.

Presently, the SCORPIO system is used for operator information and not automatic control, whereas the PRISM control engine extends this technology to cover automatic feedback control. Also, SCORPIO does not cover abnormal operation to the degree it is intended to be for PRISM. SCORPIO installation is recent, and site operational feedback has not been received. A similar system to SCORPIO called 3D MONICORE has been installed by GE in several BWRs and is well-liked by the plant operators.

7.8 Comment

The statement is made in Section 7.6.1, item 3, page 7.6-1 that the accident monitoring system provides the operator with quantitative information on the state of the plant. This is noted to be a primary protection against radiation release. Justify the safety classification of this system and its components.

Response

To quote the cited reference:

"The primary protection against radiation release is provided by several mechanisms:

1. The inherent safety features of the reactor system.
2. The Reactor Protection System which functions to limit the damage to the reactor due to accidents.
3. The accident monitoring system which provides the operators with quantitative information on the state of the plant, allows the possible initiating events to be recognized,and provides information to evaluate the incident and assure protective actions are effective and complete."

Item 3 is self-explanatory in that it performs a monitoring function only - a secondary protection function involved with accident monitoring. As to the safety classification of the system:

> Regulatory Guide 1.97 (Instrumentation for Light-water-cooled Nuclear Power Plants to assess Plant and Environs Conditions During and Following an Accident), para. 1.3.1a states: "Where the instrumentation channel signal is to be used in a computer based display, recording, and/or diagnostic program, qualification applies from the sensor to and includes the channel isolation device. The location of the isolation device should be such that it would be accessible for maintenance during accident conditions."

In full compliance with this regulatory guide, published by the U.S. Nuclear Regulatory Commission in December 1980, the PRISM Accident Monitoring System is classified as safety-related from the sensor through to and including the isolation device. The quad redundant sensors are in contact with the process being monitored. The safety-related electronics is distributed with one division in each of four Reactor Protection System Instrumentation Vaults adjacent to the reactor Head Access Area. Each vault contains a safety-related isolation device to couple to the plant optical data communication system. It should be noted that this includes a safety-related readout in each of the Reactor Protection System Vaults (accessible for maintenance during accident conditions). The plant optical data communication system and operator displays in the control room and remote shutdown facility are non-safety related.

7.9 Comment

How will the RPS sense impending sodium boiling? The flux measurement may not sense a problem, if the feedbacks hold the power steady. The temperature measurements may be late, due to sensor location and time constants. Indicate where the core outlet temperature measurements are taken (provide drawing) and the associated time constants?

Response

With the PRISM inherently safe design, natural circulation cooling and core reactivity limitations assure that boiling temperatures are never reached. See the change to Section 7.5.1.2.

The RPS will respond at sodium temperatures well below boiling.

7.10 Comment

Is there a credible means for the oil in the sodium pressure sensor to leak into the primary loop sodium?

Response

The Core High Pressure Inlet Plenum sodium pressure sensor utilizes a NaK filled section to transmit the pressure through the reactor head to the sensing elements located above the head in the Head Access Area. Any oil leakage would have to be through the double bellows of the NaK assembly, or through the many seals at the reactor head. With increasing advance in high technology sensors, we are reviewing the total sensor design and possibly a fiber optic sensor without use of bellows - oil will be used.

The pressure sensor has features to prevent oil from leaking into the reactor. The portion of the sensor assembly that contains the oil is located above the reactor closure. Were this oil container to leak, the oil would be retained above the closure immediately around the sensor and be prevented from pouring into the reactor. If the upper bellows which separates the NaK and the oil were to leak, the oil could enter the NaK capilliary tube but the oil would still be reached from the primary sodium by the capilliary tube itself.

Another important aspect of the design is that the amount of oil used is very small. If somehow all the oil leaked into the reactor, the consequences would be negligible since only a few fluid ounces are involved.

7.11 Comment

If pins in two or more assemblies fail, it will become increasingly difficult to locate the failed pins. What are GE's plans to handle such a development?

7.11 Response

When a clad failure occurs, PRISM will continue to operate in run beyond cladding breach (RBCB) mode. Pure gas leakers (generally birth defects) will be allowed to remain in the core until the next refueling outage. Breaches (gas leakers progressing to delayed neutron (DN) emitter) will remain in the core to the next refueling outage or until the cover gas fission gas or primary sodium delayed neutron activity limits are exceeded. These limits will be established in future design phases. Failed fuel will be stored in the reactor for one cycle at spent fuel storage locations.

The fuel for PRISM is expected to be very reliable with a nominal failure rate on the order of 0.2 breaches/reactor/year. Converting this into operating conditions, approximately 85% of the time the reactor will operate without any breaches in the core, 13% of the time with up to one breach and about 2% with up to two breaches in the core. This means that PRISM will operate with failed fuel very infrequently.

Gas tagging is employed for assembly location. All the pins of each fuel assembly are loaded with a unique composition of argon-neon isotopes. When the pin clad fails, the tag gas is released and travels to the cover gas where it is recovered and analyzed.

An off-line and plantwide approach is used for tag gas recovery and analysis. The gas recovery and analysis equipment is located at the radwaste building. The cover gas (with the tag) is transported from the reactor to the radwaste building using a vehicle with a storage tank, vacuum, pump, and compressor (which is part of the cover gas cleanup system). Once the tag species present in the cover gas are identified, core location of the assembly is completed by administrative procedures.

The tag gas analysis system is designed to detect up to five simultaneous pin failures.

7.12 Comment

Discuss why the plant control system (PCS) is highly unlikely to initiate an accident. Explain how the PCS is limited with respect to the number of control rods that can be withdrawn at one time.

Response

Although the Reactor Control System has yet to be developed in detail, the following concepts will be incorporated into the design to make the probability of an accident initiated by the PCS highly unlikely.

Only one of the six control rods will be withdrawn at a time. There is only one electrical power source and one PCS controller which must be commutated from one rod shim drive motor to the next. The electrical power source is insufficient to power more than one control

7.12 Response (Continued)

rod drive motor at a time. Rods will be alternatively withdrawn so that the rod positions are held within specified maximum rod-to-rod position tolerances, and within a specified tolerance from the average rod position. A limit will be placed on the time duration of single rod motion to additionally prevent continuous single rod withdrawal or unbalanced rod positions. Failures of these critêria will result in rod run-in and shutdown of the affected module.

Redundancy will be utilized in the controller to prevent most single failures or multiple unrelated failures from causing loss of Reactor Control System function. Dual shim control rod positional motors are provided on each rod, preventing single motor failure from causing loss of correct controller function.

However, if the PCS should fail such that rod run-out occurs, the RPS will trip the affected module by initiating control rod drop and activating the drive-in motor. The RPS is backed by the inherently safe design of the PRISM plant.

7.13 Comment

Is an accident in one module likely to impact on another module through the PCS? How is this likelihood minimized?

Response

Accidents in one module do not adversely impact the other modules through the PCS. If accident or anomalous conditions occur in one module, the PCS places the affected power block in the reactor modules leading turbine mode and will initiate either a slow runback to a safe power level or a fast runback and shutdown of the affected module. Turbine power reduces as the affected module power is runback.

The nearest common points of control between the modules are at the main steam header and at the feedwater pump outlet header. In the reactor modules leading turbine mode the main steam header pressure is tightly controlled by the close physical proximity to the very responsive turbine control valves. Module feedwater flows are controlled by each module's individual parameters (drum levels, steam and feedwater flow) which follow the individual modules power level and results in very little interaction between modules.

The hierarchy of control selected prevents unsafe demands from propagating through the PCS. The local controller for each reactor module will protect plant investment and override unsafe demands from the power block or plant level controls.

7.14 Comment

It seems contradictory to need a highly sophisticated control system for a relatively simple plant design. Has GE done trade studies regarding going to a very simple type control system?

7.14 Response

Several of the functions proposed for the PRISM PCS are already being used in open loop reactor control, e.g. Halden's SCORPIO and GE's 3D MONICORE. Some new technology items not in as mature a technology state, such as the Maintenance Manager and Configuration Manager functions, will be appropriately phased into the design. This will give the design the flexibility to meet the needs of the utility customers.

Difficulties anticipated in multi-module plant control by using a very simple controller will include:

- A simple type control system with a low level of plant automation would place a higher burden on the operator. Many detailed tasks would be allocated to the operator. This would lessen his ability to complete his responsibility of assuring safe, economic power block operations. For instance, if he is required to manually startup a particular reactor module, maintaining a desired heatup rate and manually performing all the actions required, his surveillance of the remaining two reactor modules, main turbine-generating system and support systems would be reduced.

- A simple type control system lacking data validation and having limited sensor and controller redundancy would be susceptible to mal-operation following single sensor or controller failures, initiating an unacceptably high number of plant disturbances.

- Inadequately designed control and information systems with limited plant state identification capabilities, can fail to properly inform operating personnel or correctly change plant control modes, contributing to increased event severity. On-line dynamic reactivity, mass balance, energy balance, vibrational and noise analysis calculations compared against anticipated operating conditions and followed by strategies for attaining a safe state during plant irregularities can contribute to reduced transient event severity.

- Rather than use a simple control system relying on operator action, an automatic control strategist function can be used to provide correct automatic response following plant upsets. For example, following a single feedwater pump trip if total reactor module power level is above the run-out capacity of the remaining feedpump (approximately 68% of rated), the reactor moldules must be runback to prevent module trips on low steam drum levels. However, should initial total reactor module power levels be below the feedwater pump capacity, no immediate action is required. Other automatic control strategist function actions inlcude runback of module power to less than bypass valve capacity following turbine trip (to conserve water inventory) and runback to house electrical load levels (~8% load) following loss of electrical grid connection (maintaining the reactor modules critical and providing stable operation at house electrical load level).

7.14 Response (Continued)

A trade study has been made on the benefits of PRISM multi-module controller automation. The report shows the high level of automation possible with the proposed control system is cost effective.

7.15 Comment

Show that the Remote Shutdown Facility (RSF) will be accessible under accident (including seismic) conditions, particularly for individuals coming from the control room. What is the maximum time that will be needed to fully man the RSF under accident conditions? What protection is needed for the operator, such as HVAC, shielding, etc.?

Response

The Remote Shutdown Facility (RSF) is in a tornado hardened, seismic category I building. Easy access to the RSF is provided from the yard and it is accessible for individuals coming from the control room. Reactor shutdown and post accident monitoring features are provided at the RSF. The control operations can be accomplished at the reactor instrument vaults by roving operators also, not dependant on individuals coming from the control room.

The RSF will provide a habitable environment under accident conditions. The calculated source term represents a relatively low value; expected doses are less than 5 REM without a safety grade HVAC system. Consequently, habitability is possible for the duration of any post accident monitoring period.

7.16 Comment

The conceptual RPS configuration is unclear with respect to the "intercom bus," Since "voting" is required, is there redundancy in this data communications?

Response

Within the RPS, safety-related redundant fiber optic inter-communication links are provided between the divisional controllers for the exchange of sensor data, timing and diagnostic information. These fiber optic intercommunications links assure divisional isolation and at the same time provide for high availability by making all RPS data available to all RPS divisional controllers.

Typically, data is transferred to an input buffer memory within each RPS division controller. When the data from all four divisions have been exchanged, each divisional controller will have four sets of identified "equivalent" data. A fully operational system will process the data as follows:

1. A division controller will ignore its own data (reserving its own observation as a spare).

7.16 Response (Continued)

2. The divisional controller will process the data from the other three divisions.

3. The divisional controller will take action based upon agreement between at least two out of the three pieces of data (a 2 out of 3 vote). Thus, if 2 out of 3 sensors indicate trip the RPS divisional controller will output a trip signal to the reactor trip logic. Reactor trip requires 2 out of the 4 divisional controllers to indicate trip, and this 2/4 division vote is done by a hard wired relay-logic circuit.

4. Should there be a division failure (bad or missing sensor data from one division or failed divisional controller), the other three divisional controllers will substitute their own "spare" data and continue with the 2 out of 3 data processing. This design permits one division to be taken out of service for maintenance, and for periodic on-line testing of each of the divisions, without affecting the 2/3 division trip and 2/4 reactor trip logic.

7.17 Comment

The description and configuration of the control rod scram latch release switching logic (Fig. 7.2-2) needs to be clarified. What are the "three sets of contacts" described in 7.2.2.2; what is the safety related boundary on Fig. 7.2-2; what is the interface between the output of the RPS (Fig. 7.2-1) and the logic on 7.2-2? Show the "circuit breaker assembly." Are the power supplies class 1E?

Response

Each of the four divisions of the RPS drives two scram breakers for each control rod latch coil. The two scram breakers for each division are physically located in each RPS division's instrument vault. The eight scram breakers are arranged in such a manner that the circuit forms a hardwired 2 out of 4 voting logic. Thus, if any two RPS divisions call for a trip, a trip will occur. One RPS division may be failed or removed from service for any reason, in any manner and will not cause an unintentional reactor trip.

Each scram breaker is envisioned as an optically coupled solid state device to provide isolation between the electronics of the RPS and the electrical power provided to the latch coil. Two different, redundant, safety-related power sources are provided for these two circuits. Each scram breaker is provided with multiple contacts. One set is used for the latch coil circuit and the additional contacts are involved in feedback for the automated diagnostic testing of the trip function.

All circuitry shown in Figure 7.2-2 is safety-related and is classified as Electrical Class 1E equipment. This includes the power sources.

7.17 Response (Continued)

The interface between the Output of the RPS (Figure 7.2-1) "To Actuator" and the logic of Figure 7.2-2 breakers, "Input From Logic" will consist of board mounted semiconducting line drivers transmitting via redundant cables to wall mounted, optically coupled, solid state scram breakers - all physically located within the supported division of the RPS instrument vault.

7.18 Comment

Explain what is meant by "The CIS logic is a system of the RPS" and "The CIS is a static system....?" Describe the logic associated with this function.

Response

Penetrations of the reactor closure that require isolation are limited to five 3" sodium processing lines and one 1-1/2" cover gas processing line. During reactor operation, these lines are closed with redundant isolation valves and prevented from opening by electrical interlocks and mechanical locking mechanisms on the valves. Strict administrative procedures also ensure the valves remain locked closed when the reactor module is in operation. These valves are located as close as practical to the reactor closure head and are within the reactor head access area (HAA). Interlocks provided by the Reactor Protection System prevent opening of these valves unless the reactor is in a shutdown mode.

The only time these valves can be opened is when the RPS system has been placed in the shutdown/maintenance mode. The interlocking associated with these valves is the total extent of the logic associated with the RPS. The automatic fire closure of the HAA is to protect investment. Appendix R, 10CFR50, is not applicable to the fire closure of the HAA. Section 7.3, Engineered Safety Feature System is deleted in this amendment.

7.19 Comment

A direct indication of position is needed for the head penetration valves. The state of the actuator is not sufficient.

Response

Agreed, we will not monitor motor control center (MCC) open/closed position or actuator signals. Any critical function valves will be monitored by position sensing devices attached directly to the valve flow control mechanical actuating mechanism.

7.20 Comment

For the EM pumps, describe the safety classification of the sensors

7.20 Response

and logic which monitor the electrical power supply. Describe the sensors and explain the indications which show the readiness of the auxiliary synchronous machines during normal operations, including the trip logic.

The EM pump, the synchronous machine, and the dual RPS breakers in the power supply lines are all safety-related and classified as electrical Class 1E equipment. The controller, the load commutated converter, and the AC power input source are not safety related.

All safety-related actions of the EM pumps and the synchronous machine are detected by the RPS through the measurement of the pump outlet pressure. Any problems with the input electrical power, the synchronous machine, or the EM pump will result in a reduction of the pump outlet pressure. Normally, the synchronous machine corrects the power factor of the EM pump. Hence, any problem that would influence the performance of a synchronous machine will degrade the efficiency of the EM pump - and be sensed as a decrease in the pump outlet pressure.

Any EM pump, synchronous machine, controller/converter or electrical power source disfunction that influences the performance of the reactor will be sensed by the RPS pressure and temperature sensors and result in a reactor trip as the safety setpoint is violated.

The electrical power supply for each EM pump is monitored at the power conditioning unit. These sensors and logic are classified as non-Class 1E.

7.21 Comment

In at least two cases, it is stated that the flow of the sodium is inferred i.e., from pump voltage and current during power and temperature across the IHX during shutdown. In both cases, it is implied as impractical to measure the flow rate directly. Please provide additional justification and the basis for the acceptability of what is provided.

Response

A direct measure of the primary flow is the preferred method of determining the flow of the primary sodium within a PRISM reactor. It is not economic to provide Class 1E sensors (four for each EM pump) that will meet the various requirements, especially the long life with the EM Pump submerged in a liquid sodium envionment.

Initially, it was proposed to infer primary sodium flow from a measurement of EM pump voltage and current. Subsequently, the RPS Setpoint Trade Study, analytically demonstrated that flow could be adequately and reliably determined from the Core Inlet High Pressure Plenum Pressure measurement. Thus, the EM pump voltage and current are not monitored as an RPS parameter.

7.22* **Comment**

Describe the method to be used to measure the air mass flow rate in the RVACS.

Response

See revised Section 7.5.2.3.

7.23* **Comment**

Section 7.9, item 4 states that following failures the operator will continue to serve in a supervisory capacity rather than assume manual control of the affected system. A table of how the affected system will function or be controlled would be helpful.

Response

See added Table 7.9-1.

7.24* **Comment**

We understand that it is no longer true that the RPS signal for primary sodium flow rate is inferred from EM pump voltage and current. The PSID should be updated.

Response

The measurement method for primary sodium core flow is now indicated in Section 7.2-1.

7.25* **Comment**

Describe the Loose Parts Monitoring System planned for PRISM.

Response

The loose parts monitoring system has not been defined beyond the description provided in Section 7.6-7 at this conceptual design stage.

7.26* **Comment**

Update the list of Regulatory Guides to include RG 1.151, 1.152, and 1.153, or justify their exclusion. Justify the exclusion of RG 1.105.

Response

See the change to Section 7.1.2.

RG 1.151, "Instrument Sensing Lines," July 1983 is specifically excluded because the RPS does not have any instrument sensing lines.

*Comments requiring textual improvement.

7.27* **Comment**

Provide the safety design bases for initiating an EM Pump coastdown or reactor trip. In Chapter 8, the discussion only mentions electrical disturbance related trips. Please clarify.

Response

See the change to Section 7.4.

7.28* **Comment**

Show the safety-related boundary of Figure 7.2-3. Is the power Class 1E?

Response

See Figure 7.2-3. The safety related boundary is also indicated on Figures 7.2-1a, 7.2-1b and 7.2-2.

7.29* **Comment**

Safety-related manual scram capability is provided locally. Describe the information used by the operator to perform such manual functions and the communication links to the central control room.

Response

The information needed by the operator to perform a manual scram is now provided in Section 7.2-1.

7.30** **Comment**

Analysis of 316 stainless steel samples for impurities has been noted on page 7.6-3 of the PSID. Material sampling should be considered and described further in the development of the QA program.

Response

Material sampling is defined in the materials section of Chapter 1 and in the QA program of Chapter 17.

7.31** **Comment**

The self-limiting characteristics of the metal fuel have been noted several times. The manner in which this will be demonstrated should be described further.

* Comments requiring textual improvement.
**Comments for further consideration.

7.31 Response

See Chapter 4.

7.32** Comment

Descriptions of the conceptual RPS incude isolation devices. There are different schools of thought on testing and qualification of isolation devices and we would recommend close communication with the staff as the RPS is developed.

Response

The PRISM design team will maintain interactions with the NRC staff as the RPS design matures. Special attention will be given to the testing and qualification of the isolation devices between the independent divisions.

7.33** Comment

The staff is of the opinion that the concept of the operator as a non-safety feature is not acceptable at the present time. We do note that there is redundant capability in the various "roving" operators. As this concept is developed further, we would recommend definition of the number of available (roving) operators, their typical locations, functions, training and licensing requirements if they are to perform a shutdown or monitoring function. In addition, while extensive provisions have been made for local control, the vaults, as described, do not contain communication paths with the central control room. Communication should be provided in a reliable manner. Also, performance of the operators' functions under adverse conditions will require adequate environmental conditions such as shielding and air supply.

Response

GE understands the staff's position to be that operators on any nuclear site have generic safety functions. It is our further understanding that these generic safety functions are related to monitoring and communicating during emergency conditions and if the plant emergency requires, taking actions to improve the site safety status. It has also been noted by the staff that in the case of PRISM there is not a specific location on the standard plant site where operations staff can be expected to be fully protected from accident or environmental hazards.

We can concur with the staff's position that there are generic safety functions that operations staff can and should perform during emergencies on any nuclear site, as long as the definition of safety functions does not mean: 1) assuring accident releases are less

**Comments for further consideration.

7.33** **Response (Continued)**

than 10CFR100, 2) assuring integrity of primary coolant boundary during accidents, or 3) assuring reactor shutdown during accidents. These three functions are the bases for design of safety-related equipment and, in the case of electrical hardware, qualification as 1E. In the case of PRISM these three safety functions are assured through automatic action of the safety-related Reactor Protection System (RPS) and inherent features in the design. The inherent features include those that prevent initiation of accidents as well as the inherent mitigation of extremely low probability events such as ATWS. Information has been provided that supports the fact that the facility operations staff have no functions during the Design Basis Accidents (DBA) and a broad spectrum of Beyond Design Basis Accidents (BDBA) other than monitoring and communicating. It must be acknowledged of course that accidents such as fires and chemical spills as well as environmental hazards such as seismic events and tornados, could demand appropriate operations staff emergency response. These actions would contribute to reducing the owner's investment loss. To date we have not identified any accidents that would require the operations staff to act to assure that off-site radiological consequences are less than 10CFR100 limits. However, we concur on the need to provide reliable on-site and off-site communication systems and have established requirements to do this.

The final detailing of operations staff emergency tasks will not be completed until the detail design and task analysis is done. It has been and continues to be an objective of the PRISM design approach to eliminate all safety-related operator responses during an emergency. That is, the operations staff does not need to interact with any safety-related equipment to assure safety-related functions are performed. Operators can and will perform the generic safety functions associated with protecting the investment and this can have a beneficial impact on the safety status of the plant during emergencies.

With this background in mind, it is acknowledged that there is an advantage to assuring that operations personnel will be available on site to perform the generic safety functions identified by the NRC staff. The auxiliary shutdown console located in the Reactor Service Building (RSB) will therefore be equipped such that appropriately trained operations staff can access and remain at this location during major natural environmental events or accident caused environments such as smoke or other noxious airborne contamination. From this location all nine reactors can be monitored, and a scram of any of the reactors can be manually initiated. The auxiliary shutdown console is not qualified Class 1E. At this stage of design, it is expected that the staff normally working in the RSB will have a role associated with the auxiliary shutdown and monitoring station in the RSB but staffing details have not been determined. The remote shutdown panels located in the instrument vaults at each module will be retained and designed to meet 10CRF50 requirements for remote shutdown and monitoring equipment outside the control room. These panels are qualified Class 1E.

**Comments for further consideration.

7.33** <u>Response</u> (Continued)

In Summary:

1. The Control Room provides a facility from which the operator performs all the monitoring and control tasks during normal and abnormal conditions as long as the Plant Control System is operative and the control room is available.

2. The Auxiliary Shutdown console in the Reactor Service Building provides an environmentally protected area from which the operations staff can initiate manual scram and monitor reactor shutdown for all plant events and conditions except those highly unlikely cases where the Plant Control System has failed in a way that prevents this action.

3. The nine (one per reactor) instrument vaults are seismically protected locations that house Class IE qualified equipment. From each Vault an operator can initiate a manual Class IE scram (the only shutdown action required) and perform post-accident monitoring of that specific reactor.

4. Reliable (and redundant) inter-communication channels are provided between all three major operator interface areas - the Control Room, the Auxiliary Shutdown Console in the Reactor Service Building, and the Instrument Vaults.

**Comments for further consideration.

8

RESPONSES TO NRC COMMENTS ON PRISM PSID CHAPTER 8

8.1 Comment

Is there any feasible way to reverse the voltage across the EM pumps?

Response

The question seems to address the possibility of reversing the phase sequences in the EM pump power supply circuit. This possibility is very remote. The EM pump is supplied by very large cables which run in steel conduits. To reverse phase sequences, these cables would have to be disconnected at the terminals and reconnected incorrectly. This will be virtually impossible because of the physical characteristics of very stiff cables. This could be postulated only during a repair or maintenance operation. However, test procedures would discover the mistake prior to operation of the EM pump.

8.2 Comment

Provide a description of the effects of a (1) loss of any Class 1E plant electrical bus and (2) loss of any non-Class 1E bus.

Response

(1) No effects. Loss of a Class 1E plant electrical bus will result in loss of power to those loads which are connected to the bus. However, the PRISM design provides redundant and separated power sources for all safety-related components. Hence, the loss of a Class 1E plant electrical bus will have no safety-related consequence.

(2) Loss of a non-Class 1E bus will result in loss of power to those loads which are connected to that bus. The loss of a 7.2KV-AC distribution bus supplying the EM pumps will result in a safety-related reactor trip initiated by the RPS as a response to a "loss of flow." The Plant Control System and its critical components are supplied by battery-backed uninterruptible power sources.

8.3 Comment

Justify the exclusion of the power conditioning unit and the load commutated inverter (LCI) from the Class 1E boundary.

Response

The external electrical power input to the EM pumps provides primary flow within the reactor. The safety-related action is the sudden loss of this external power (a loss of flow event) - see the answer to Comment 7.21. The sudden loss of power is a safety-related event protected by the safety-related synchronous machine. In the event of

8.3 <u>Response</u> (Continued)

a loss of primary power to the EM pumps, the synchronous machine provides the energy necessary to generate the safety-related flow coastdown. Thus, the primary source of power, the EM pump controller and the load commutated converter (LCI) do not perform a safety-related function so are not classified as electrical Class 1E equipment.

The EM pump and synchronous machines are necessary to the operation of a PRISM reactor. Likewise, the reactor cannot operate without a power source. It is not economically feasible to provide sufficient battery power to operate the pumps during a power outage. Therefore the pumps are powered from non-safety-related power sources. If the power should fail, the RPS will sense an immediate loss of pressure and execute a reactor trip protecting the reactor, etc. From an economic viewpoint, the less equipment that must be classified as safety-related without compromising safety is desirable. Hence, the PRISM approach is to classify only those components necessary as safety-related. It is not necessary to classify the power source and controller/commutator as safety-related. These components/sources are isolated by the double safty-related breakers of the RPS.

The power conditioning unit (including the load commutated inverter) will be classified as non-Class 1E. The PCU supplies power during various plant conditions, but has no safety related function. Consequently, exclusion of the PCU from the Class 1E boundary is justified for the coastdown of the EM pump; the stored kinetic energy in the synchronous machine is utilized.

8.4* <u>Comment</u>

Please provide drawings that show the cables that run between the synchronous machines and the EM pumps. Show that (1) the machines and (2) the cables are properly separated, so that at least two out of four will always be available.

<u>Response</u>

See Figure 8.3-b, EM Pump Coastdown Power Cable separation.

8.5* <u>Comment</u>

Provide a listing (similar to Chapter 7) of the appropriate Regulatory Guides, Standards, etc., for Chapter 8.

<u>Response</u>

The Regulatory Guides applicable to the PRISM design are listed in Table 1.8-1 of the PSID. A more detailed list of applicable Regulatory Guides and industry standards will be provided during the detailed design process.

*Comments requiring textual improvement.

8.6* Comment

Are there any non-Class 1E loads on the Class 1E buses?

Response

See Section 8.3.1.

8.7* Comment

Are the 4 batteries shown on Figures 8.3-2, 8.3-3 and 8.3-4 separate batteries?

Response

Yes, one battery set for each channel.

8.8** Comment

Per the requirements of 10 CFR 73.55(c)(5), all exterior areas within the protected areas are to be provided with illumination not less than 0.2 footcandles at ground level. The illumination is to be supplied by an uninterruptible power source.

Response

Illumination for the protected area will be backed up by an uninterruptible power source.

* Comments requiring textual improvement.
**Comments for further consideration.

9

9.1 Comment

Assuming a sodium fire begins, what areas are susceptible to the spreading of a sodium aerosol before the fire can be extinguished? In particular, is it possible to compromise all EM pump synchronous converters simultaneously?

Response

Sodium fires could originate in the steam generator silo, IHTS piping tunnel, or possibly the primary cold trap vault if the operator neglects to freeze the system prior to opening the system for component replacement. In all cases sodium catch pans and fire suppression decks will prevent continued pool burning and limit the burning to less than 10% of the spill. In addition, the plant will be shutdown in response to a continued indication of a major sodium fire. The aerosol which is released to the atmosphere will be available for ingestion into the various HVAC building ventilation inlets assuming the wind carries the sodium aerosol in their direction.

The reactor facility areas, such as the Head Access Area (HAA) above the reactor deck and the close coupled 1E vaults containing the safety related reactor shutdown electronics and the synchronous coastdown machines, contain instrumentation (sodium aerosol and radiation detectors) which will cause the HVACS to revert to a sealed system cooling mode by closing the fire dampers in the HVAC ducts. Thus, the possibility of a sodium fire impacting the operation of a synchronous machine is quite remote. In addition, since the reactor and its associated EM pumps will not be operated without the synchronous machines (they provide a necessary reactive load and a module shutdown occurs if any one of the machines fails) it is extremely unlikely that more than one machine would fail to provide the 140 second long coastdown flow as the units would have to fail at same time.

9.2 Comment

Discuss provisions and criteria for monitoring water buildup and removing such water within the reactor silo.

Response

The exterior concrete surfaces of the silo walls are waterproofed and water stops are provided at construction joints so that water leakage into the silo is not contemplated. Rain protection is provided at the RVACS inlets and outlets. In the unlikely event that water leakage does occur, it will evaporate and be removed by the RVACS hot air exhaust. Under normal conditions, the lower end of the reactor module exterior surface operates at temperatures between 300°F and 400°F. This hot surface is a heat source which together with the RVACS air stream will evaporate any water on the silo floor. It is estimated

(9.2 Response - continued)

that these mechanisms can evaporate and carry away an average of 40,000 gallons of water, either rain or water seepage, during each day of plant operation. To confirm normal conditions, three redundant water detectors are located at the bottom of the RVACS duct for continuous monitoring. The reactor silo walls and floor will be periodically examined by remote visual means.

9.3* Comment

The text on page 9.5-4, last paragraph, would be cleaner if "...except as necessary." was replaced with "...except in the area of the steam generators."

Response

The clarification change has been made.

*Comments for textual improvement.

10

10.1 Comment

Regarding the blowdown from the steam generator drums mentioned on Page 10.1.1 - what is the flowrate and temperature (saturated water?) of this flow at full power conditions? How does the flowrate vary at part-load conditions? (Note: this information may be needed for calculations.)

Response

The blowdown flow rate from the steam generator drum is two percent of the normal steam flow or 38,000 lb/hr. The drum blowdown is saturated water at 543°F and 990 psia. The drum blowdown rate is held constant and does not vary with reduced load.

10.2 Comment

Most light water reactor (LWR) transients begin in the balance of plant. As the PRISM balance of plant is quite similar to an LWR, a similar frequency of transient initiators can be anticipated, and these initiators would impact on three reactor modules concurrently. Has this been factored into the PRISM duty cycle, PRA and availability estimates? Response to this question should be coordinated with related question, 5.13, and future questions regarding the PRA for PRISM.

Response

BOP-initiated transients have been factored into the PRISM duty cycle, PRA, and availability estimates. Specifically, each PRISM module has been assumed to be challenged by all BOP transients with proper account for: (1) the response of other modules to each transient, (2) any synergistic effects which may result from this response. It should be noted that these synergistic effects have been reduced to a negligible degree as a result of a) physical separation of the modules, b) capabilty for controlling each module individually, and c) as discussed in the response to Comment 5-13, the capability of a PRISM power block to cope with the tripping of one module without interruption of power production.

The PRISM duty cycle conservatively defines a large number of events (>10 events/year/module) which exceeds the corresponding number in current LWR operating experience (roughly 5 scrams/average reactor/year). Some of the duty cycle scram and fast runback events defined in Appendix D are "generic" and do not define the location of the initiator. Depending on the assumed location of the initiator, the percentage of duty cycle events originating in the BOP could be between 34% and 91% of the total events. This is consistent with the fraction of 70% attributed to LWR BOP originated events reported in NUREG/CR-4783.

(10.2 Response - continued)

BOP-initiated transients have been included in the PRISM PRA. As shown in Appendix A, Section A4.1.2 of the PSID, the PRA includes the BOP-initiated transients of: loss of operating power heat removal, loss of shutdown heat removal via BOP, station blackout, spurious scram, and forced shutdown. The frequencies of spurious scram and forced outages for PRISM are expected to be less than those experienced in currently operating power plants due to the improved capabilities of the PRISM control system for 1) signal validation and interpretation, 2) fault tolerance, and 3) proper control of the system to avoid unnecessary shutdowns.

11

12

13

14.1 Comment

A test of the Auxiliary Cooling System (ACS) should be included, as credit is taken for this component in the PRA.

Response

The ACS is a non-safety grade system designed to high industrial standards whose function under existing plant design is accounted for in the PRA. ACS testing will consist of standard preoperational tests, and depending on the prototype test facility options, startup tests will be conducted when the steam generator construction is completed.

14.2 Comment

A test should be added to make sure that the control rods can't "float up" during refueling if the pumps are erroneously turned to 100% flow.

Response

The movable absorber bundles are designed with sufficient weight to prevent uplift at full flow conditions. This capability will be tested out-of-reactor in a development flow and pressure drop test program to assure that all assemblies behave hydraulically as predicted. This testing would not be part of the Safety Test Program.

14.3 Comment

A test of the Sodium Water Reaction Protection Relief System (SWRPRS) should be included in the test matrix to demonstrate its effectiveness in maintaining the integrity of the IHX (i.e., the primary system pressure boundary) during sodium water reaction events.

Response

A test of SWRPRS as part of the Safety Test is not practical or needed. The key component for operation of SWRPRS during a large sodium-water event is the double rupture disks, which must rupture at the prescribed pressure rating. Such destructive tests will be run by the rupture disk manufacturer as part of his development/verification program. On the Clinch River program, for example, nine of ten rupture disks burst within specifications. The tenth burst at a somewhat lower pressure.

In accordance with ASME Code requirements, pressure tests on the IHX will be run by the manufacturer to verify its ability to withstand the design pressures. The shell side of IHX will be pressure tested at 1.25 times the primary side design pressure of 20 psig. The tube side will be tested at 1.25 times the intermediate side pressure of 250 psig, the SWRPRS rupture disk bursting pressure.

Response (continued)

In addition, the Steam Generator Large Leak Tests conducted at ETEC in support of Clinch River have validated the analysis methods and codes to predict: (1) the magnitude and growth of medium to large sodium-water reactions, (2) the bursting dynamics of the double rupture disks, and (3) the transient flows and pressure in the secondary sodium loop and the SWRPRS following disk rupture.

14.4 Comment

A test matrix with some EM pump coastdown variation should be included to analyze the margin to boiling relative to the EM pump coastdown.

Response

The Safety Test Program includes bootstrapped testing of loss of programmed flow coastdown to assure the predicted margins to boiling. The program would only test multiple pump failures from power levels that allow further module operations after the tests.

For example, one instantaneous pump failure (loss of coastdown) is tested from 100% power, two simultaneous pump failures is tested from 80% power, three simultaneous failures may only be tested from 50% power and the instantaneous failure of all four pumps is tested at 25% power. The test program limitations for 2 or more pump failures will preserve reactor life for future operations and tests. The reactor is designed to accommodate loss of 2 pump failures at 100% power but would leave the module incapable of immediate restart and unlimited operation.

14.5 Comment

A test matrix for the seismic isolators should be developed. This should include displace tests of their ability to return back to the proper alignment. Suggested tests include:

- Placing the vessel under torque to determine its response at the seismic isolators.
- If the isolators fail to return the vessel to its original position, is it possible to block the RVACS.
- Can this displacement test be used to test the 30" gimball in the IHTS and its ability to follow the reactor system during a seismic event?

Response

Testing of the seismic isolator is an element of the PRISM development tasks listed in Table 1.5-1 and seismic response tests, as discussed

Response (continued)

in Section 14.2.2.2, are being considered for the prototype module. For example, a horizontal displacement can be imposed on the seismic isolated assembly of the module and released to demonstrate performance. A 130-ton load would provide such a displacement. The RVACS entire heated flow path from the inlet plenum at the bottom of the silo to the exit is part of the seismic isolated assembly and would not be significantly affected by the displacement. The margin in the cold flow path permits 80% blockage with adequate inlet plenum air. Thus, the assembly displacement would not be a significant restriction to the cold flow.

The gimball bellows are mainly needed for the initial heatup of the IHTS and could be an element of the seismic tests. However, the definition of the tests is continuing as the design matures.

14.6 Comment

During the transient overpower tests (TOP), only $0.22 reactivity has been given as an initiator. This should be expanded, perhaps to values near $0.50, in order to demonstrate the inherent safety of this plant to more serious reactivity insertions. Assuming the PRISM module could perform acceptably, this would provide improved documentation of the margins for the certification process.

Response

The unprotected overpower tests will bootstrap up to the maximum credible reactivity insertion, limited only by the need to not damage the module to the extent that future tests are precluded.

The magnitude of the insertion due to a control rod runout varies with the core design. It is defined to be the insertion resulting from a single rod runout. The magnitude is thus the sum of one rod's contribution to burnup reactivity swing and excess reactivity suppression (due to uncertainties stackup) multiplied by the interaction factor.

The PSID, Revision 1, Amendment 1 (GEFR-00793) documents a "zero" burnup swing core. Table 4.3-9 shows the burnup swing to be 0.06¢ and the statistically combined uncertainties to be 0.31$ for an operational n^{th} core. Thus for this core, the reactivity insertion from a single rod runout from full power is (0.06$ + 0.31$)/6, or 0.06$.

The magnitudes of uncertainties will be investigated further in FY88 tasks. Recent FY87 work on reactivity uncertainties yields results very similar to that documented in the PSID Amendment 1. Following is a discussion of each of the uncertainty elements.

Response (continued)

Control Worth Requirement and Uncertainties

The core reactivity worth uncertainties are defined in Table F7-1. During conceptual design, large uncertainties apply to calculational-based uncertainties because the codes have not been calibrated by operational experience with a prototypical core. These large uncertainties must be included in the estimation of the control system worth requirement so that the absorber system will be designed with sufficient worth to guarantee a sufficiently subcritical core at shutdown with limited prior knowledge. However, for estimation of the excess reactivity to be designed into replicate cores following code calibration by the Safety Test and prototypic operation, lesser calculational uncertainties are appropriate. For PRISM, it is assumed that the calculational-based uncertainties will be reduced to magnitudes shown in Table F14-6.1.

The uncertainty in calculating the temperature defect is assumed to be 20% of the defect magnitude for control worth requirement specification, or 0.24\$. It is assumed that the magnitude of the uncertainty will be reduced with code calibration to the worth of the core radial feedback from potential inelastic bowing and gap closure. From CORTAC analyses, this reactivity effect is estimated to be 0.10\$.

The calculational uncertainty in the criticality prediction is assumed (based upon CRBR methods) to be 1.00\$ for control worth requirement specification. It is assumed that code calibration from the prototype core will reduce this calculational uncertainty to 0.10\$.

The refueling calculational uncertainty is caused by the use of batch averaged nuclear analysis to represent discreet fuel management and refueling. For specification of the worth requirement, the uncertainty is assumed to be 1.00\$, based upon CRBR precedent. It is assumed that code calibration and nuclear analysis with discrete fuel representation will reduce this calculational uncertainty to 0.10\$.

The calculation of the burnup swing has an assumed uncertainty of 15% of the reactivity swing, or 0.01\$ for the reference core. It is assumed that the reload core batches will be specified to yield a zero burnup swing, but that the uncertainty in the related computations will remain 0.01\$.

The uncertainty in the reactivity worth of the fuel reload is determined by the tolerance on the fissile content of the fuel. PRISM fuel is specified with a fissile fabrication tolerance of $\pm$0.25% of heavy metal. The reactivity worth of this fissile tolerance is $\pm$0.15\$, yielding an uncertainty magnitude (peak to peak span) of 0.30\$. (Note that since the fissile fraction in PRISM fuel is about 25%, this fuel fabrication tolerance is basically equivalent to 1% relative tolerancing, which is the same as is used in conventional oxide LMR fuel fabrication.)

Response (continued)

The uncertainties are assumed statistically independent and thus are combined as the square root of the sum of the squares as shown in Table F14-6.1. The total uncertainty to be included in the control system reactivity worth requirement calculation is 1.69\$, while the uncertainty in an operating n^{th} core reactivity is 0.35\$.

Table F14-6.1
Reactivity Uncertainties

UNCERTAINTY ELEMENT	UNCERTAINTY MAGNITUDE (\$)	
	FOR WORTH REQUIREMENT	FOR Nth CORE OPERATIONAL REACTIVITY
CALCULATIONAL UNCERTAINTIES:		
Temperature Defect	0.24	0.10
Criticality Prediction	1.00	0.10
Refueling Modelling	1.00	0.10
Burnup Swing	0.01	0.01
PHYSICAL UNCERTAINTIES:		
Fissile Loading	0.90	0.30
COMBINED UNCERTAINTY:	1.69	0.35

14.7 Comment

The multiple failure tests indicate that they will start out at 60% power. This would be better if the tests could start at 100% power. Please indicate where this would be possible.

Response

The multiple failure tests are intended to demonstrate the ability of the plant to accommodate extremely severe events which have a probability so low that these events are "once-in-a-plant-lifetime" events. As such the design philosophy is that the plant is able to withstand such events without endangering public health or safety. However, it is assumed that if such events occurred in an actual operating plant extensive testing and repair of the plant may be required before bringing the plant back into service, and in some severe cases it may not be possible to restore the plant to operational status. Since several "once-in-a-plant-lifetime" events will be imposed on the PRISM safety test facility, it would be inadvisable to run all of these severe "once-in-a-plant-lifetime" tests at 100% power. Even if the relatively less severe tests were run at 60% power levels and the final, most severe test was run from 100% power, the effect may be to reduce the overall useful plant lifetime. In addition to running bounding safety tests, another purpose of the PRISM safety test facility is to demonstrate the ability to monitor the various reactivity parameters over the entire plant lifetime. Therefore, it would be impractical to run a series of tests of such severity that the available plant lifetime and therefore the goal of demonstrating lifetime monitoring would be compromised. The 60% power level for these severe tests has been chosen such that the power level is sufficient to provide data for a valid extrapolation to 100% power by demonstrating the physical concepts involved. At the same time the 60% power level is sufficiently low to allow these tests to continue for extended periods of time into the various accident scenarios without significantly shortening plant life.

14.8 Comment

Design Certification requires that the prototype plant behavior be completely understood. The following comments apply to instrumentation needed for such characterization.

- Flux detectors are needed in-vessel as well as ex-vessel to supply benchmarking data. This plant's neutronic characteristics must be well understood to extrapolate to other accidents, and other fluid dynamic events not covered in the test plant.

    - The flux shape during transients will also supply the safety analyst with modeling insights.

14.8 <u>Comment</u> (continued)

. Without detailed flux and power distributions, the inherent safety mechanism will not be fully understood.

- Detectors to measure bowing during the tests are needed to fully understand this inherent safety feature.

- As many channels as possible should have a temperature, pressure, and flow probes to map the state of the core.

<u>Response</u>

The PRISM safety test facility is to be built as a prototype for certification and as such an overriding consideration is that both the plant and its instrumentation be as prototypic as possible. Additional instrumentation will be used to gain additional insight regarding the details of the plant response to a variety of conditions including accident conditions. However, the addition of instrumentation must be limited to that which will not compromise the prototypicality of the plant.

Flux detectors in-vessel as well as ex-vessel to supply benchmark data will be limited to that which will not compromise the prototypicality of the plant. While such concerns may be valid for large water reactors, the PRISM plant's neutronic characteristics of a small, tightly coupled reactor reduce the need for such information. Data available from the core physics testing performed during initial criticality will verify the degree of coupling present during steady state and some transient conditions. If additional information for abnormal conditions such as sodium voiding are needed, these data can be obtained using mock-ups of the core in zero power facilities such as ZPPR. Also, integrated flux shapes during transients can be obtained by the use of retrievable flux wires and coupons as is done in the TREAT reactor.

The use of detectors to measure bowing during the tests has been discussed and appears to be impractical. Consideration has also been given to measuring bowing using dummy subassemblies during hot functinal testing. Both laser measured distortion and strain gauge type measurements are under consideration but these also appear difficult to implement.

Using a temperature, pressure, and flow probe in almost every channel would add significant complexity to the reactor upper internal structure. This would both increase the cost and have an adverse effect on maintaining prototypicality in the reactor. The probe alone would distort the flow patterns. As an alternative, better information can be obtained by the use of a few selected, special, instrumented subassemblies such as those that have been used in EBR-II. Temperature probes typically only provide information of the

<u>Response</u> (continued)

outlet conditions of a subassembly or conditions in the upper portion of the subassembly. By using instrumented subassemblies it is possible to obtain detailed information along the entire length of selected subassemblies. Also wire wrap thermocouples used in instrumented subassemblies have no effect on the flow patterns within the subassemblies as would the intrusion of a probe. The temperature information can be used to determine the axial power distribution within the assembly. The flux distribution can then be calculated from the power distribution and steady state neutronics data.

14.9 <u>Comment</u>

The reactor system should be analyzed for several differenct levels of burnup during a fuel cycle due to changes in the reactivity feedback effects with burnup.

> Determination of the differences in the feedback characteristics of the fuel as the system goes from a U-Zr based fuel to a U-Zr-Pu based fuel in the equilibrium cycle should be included in the test plan matrix.
>
> Fuel behavior characteristics such as fuel slumping, low melting point eutectic formation, changes in the axial porosity, wastage of the clad, Zr migration and others should be studied.

<u>Response</u>

The core recommended for PRISM has a "zero" burnup swing. The core reactivity state does not change appreciably with burnup, thus interim tests at various burnups will not yield new data.

The Safety Test Program will use U-Pu-Zr fuel, not U-Zr fuel. Thus, for certification, the prototypic core and fuel will be used. No plans are presently under consideration for certification of a U-Pu-Zr fuel cycle core with a U-Zr test.

The effects of fuel burnup are to be investigated in the metal fuel development programs at EBR-II, FFTF, TREAT, etc. These effects will not require further testing in the Safety Test Program. The extended module operation after completion of the Safety Test Program will address operability, maintainability and reliability issues.

14.10 <u>Comment</u>

Regarding the Reactor Vessel Hydraulic Tests (water) at ANL, can a transient in which all power to one of the EM pumps is cut off instantaneously be simulated by this unit? With failure to scram?

<u>Response</u>

The capability for performing this type of transient test in the ANL water facility currently exists only for low power, low flow (less than 10 percent) conditions. In fact, a transient similar to it was performed during 1987 and the results are currently being evaluated by ANL. Reversed flow in the shutdown pump was observed in this test. The capability to perform the test at full power without scram does not currently exist. The proposed test was discussed with ANL and it is not known if the facility can be modified for this purpose because a relatively complex valving and control system would be required. This aspect could be explored further by ANL if desired.

14.11 <u>Comment</u>

As part of the prototype test program, would GE be willing to simulate loss of all power to one of the EM pumps (with failure to scram)?

<u>Response</u>

See the response to Comment 14.4.

14.12 <u>Comment</u>

Experience with light water reactors has shown that plant characteristics and hardware change with plant age. Will provisions be made for periodic testing over the life time of a plant, or the prototype should it continue to operate, to characterize these changes and insure that the plant is operating within the understanding used in the certification process?

<u>Response</u>

Yes, a simple set of criteria and associated monitorable parameters will be developed in the prototype reactor module safety tests that characterizes the reactor inherent response explored by the tests. There are five parameters, the net (power flow) decrement, power/flow coefficient, inlet temperature coefficient, transient overpower initiator, and flow coastdown time that provide inherent control of reactivity. These parameters will form the basis for Technical Specification monitoring and the continuing demonstration of satisfactory inherent response. Reactivity coefficients have been measured in EBR-II and a program is in place to develop on-line monitoring for assuring inherent control of reactivity.

14.13 Comment

Discuss the potential for sodium freezing and the consequences.

Response

The design of the reactor system and associated RVACS will be such that the probability of sodium freezing at the bottom of the vessel is minimal. An analysis of the potential for freezing in the current design was performed. Results of this analysis show that the sodium temperature will not drop below 320°F during a lengthy refueling outage with a hot pool temperature of 400°F. Thus, there is adequate margin assuming that the sodium purity is such that freezing occurs at about 270°F. Additional parametric evaluations are planned to look at the effect of RVACS air inlet temperatures to -20°F, for which the potential for sodium freezing will be greater. The result of this study will suggest whether it is necessary to extend the cylindrical flow baffle surrounding the fixed radial shielding completely to the bottom of the vessel (instead of ending at the approximate elevation of the inlet plenum). This modified flow arrangement will result in complete sweep out of all cold sodium in the bottom vessel head region with an acceptable reduction in natural convection thermal head for core cooling. There also appears to be operational procedures such as starting the primary EM pumps at intervals during the refueling operation to sweep out and mix the sodium in the lower plenum region that could be implemented. The consequences of sodium freezing in case these measures fail have not been analyzed except that they would be undesirable and should be avoided.

14.14 Comment

Discuss the potential for penetrations in the RVACS collector, the effect of penetrations, safety limits for penetration size and propose tests, whether in the prototype facility or a separate effects facility.

Response

The meaning of this question is uncertain but is understood to mean either 1) penetrations in the collector cylinder for instrumentation, in-service inspection, cleanup etc., 2) openings at various elevations to allow RVACS cooling air to partially by-pass the normal air flow path to reduce the effect of blockage at the bottom of the reactor silo or 3) openings created by accident conditions.

In response to item 1 above GE is not aware that any penetration in the collector cylinder is required. In response to item 2, i.e., provide openings in the collector cylinder to reduce the effect of a postulated air flow blockage, it is noted that this feature is not a part of the current PRISM design. If it should be included at a later time, the performance of the RVACS would have to be studied carefully both analytically and experimentally. No thought has been given to the experimental approach, but it is visualized that the NSTF facility at ANL (full-scale, segment air-side model test) could be modified to enable such tests to be conducted. Item 3 has not been analyzed to date. However it can be studied similar to item 2.

RESPONSES TO NRC COMMENTS
ON PSID CHAPTER 15

15.1 Comment

In Section 15.1, it is explained that the PRA was used to separate DBEs from BDBEs, and thus it eliminates several improbable events from Chapter 15 analysis. Given that there is greater uncertainty regarding system and component reliability for new reactor types, discuss the steps that have been taken to assure consideration of marginally improbable events.

Response

The marginally improbable events are considered as beyond design basis events and are analyzed in Appendix E of the PSID. It is shown in Appendix E that these events (unprotected loss of primary flow, unprotected loss of heat sink, and one-rod transient overpower) will not result in fuel melting, cladding breach or local sodium boiling. Therefore, there are very large margins for public safety with the marginally improbable events. It is intended that these events will be run in the PRISM safety test and the margins demonstrated thereby.

15.2 Comment

In Table 15.3-3, for "Unlikely" and "Extremely Unlikely" events, the peak and long term (1450°F and 1340°F, respectively) are in the same range as the eutectic temperature (1340°F). As eutectic formation is a time dependent process, a brief violation of the threshold temperature is probably acceptable. How is the time-dependent consideration factored into the decision making process (i.e., in setting acceptance criteria)?

Response

The lowest temperature for fuel-clad eutectic formation has been determined experimentally by Argonne National Laboratory to be 1290F. That is, below 1290°F a eutectic does not form and there is, therefore, no attack of the cladding. The rate of cladding attack by eutectic increases with temperature above 1290°F. As shown in the attached Figure F15.2-1, at 1340°F, 2.8 hours are required to penetrate 10 mils of cladding (about one-half the cladding thickness). At 1500°F, about one-half hour is required to penetrate 10 mils of cladding.

Therefore, a long-term (>300 sec) limit of 1290°F has been set for the fuel-cladding interface in addition to the 1450°F short-term cladding limit (based on cladding strain). Interface temperatures of up to 1450°F (the cladding short-term limit) for 300 sec or less will result in penetrations of less than 0.5 mils of cladding and are completely acceptable.

15.3 Comment

On page 15.4-3, where the ARIES-P model is described, a "pump pony-motor" drive speed is mentioned. Please discuss this and any credit taken for it in the accident analyses or the PRA.

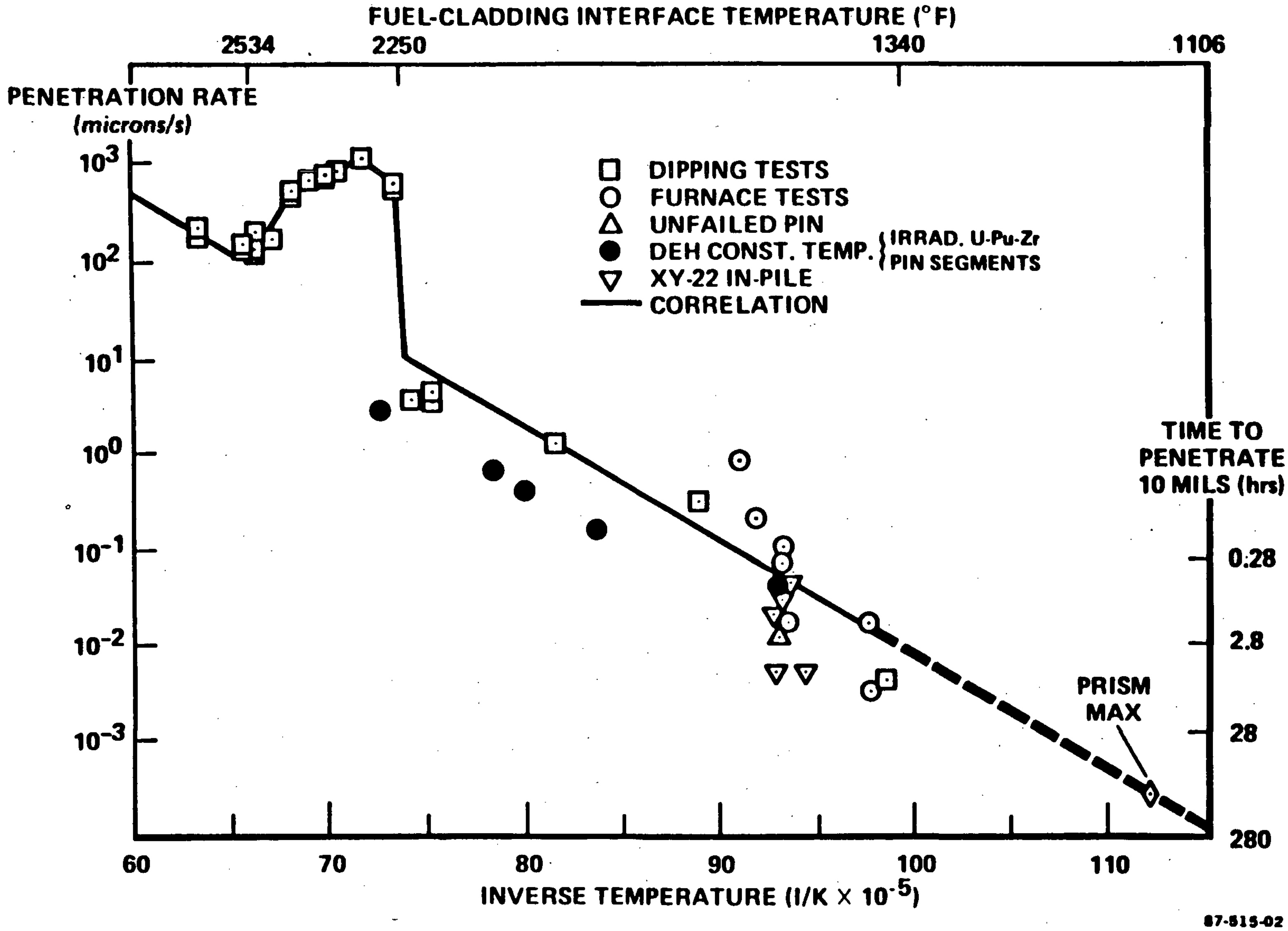


Figure 15.2–1 RATE OF CLADDING PENETRATION BY URANIUM–BASED FUELS

Response

The pump pony-motor referred to on page 15.4-3 is the pony-motor drive for the intermediate sodium pump. The pony motor is designed to deliver ten percent of rated IHTS sodium flow and engages at the preselected speed of 175 rpm as the main motor coasts down following a trip. The pony motor is an integral part of the IHTS, and is included in the ARIES-P model and in the PRA analysis.

15.4 Comment

A whole series of likely transients are not considered in Chapter 15: pump trips, loss-of-power, turbine trips, small leak in steam generator, postulated breakage of one of the cables to the EM pumps, etc. It is likely that these are probably not major challenges to the safety of PRISM, but some cursory analysis should be provided. At a minimum, provide a list of those events considered and analyzed and justify restricting Chapter 15 to the limiting cases.

Response

Recently, four extreme events that envelope related duty-cycle events were analyzed. The four bounding events were:

- Instantaneous loss of IHTS in all three modules
- Instantaneous loss of steam generator steam/water inventory in all three modules
- Isolation of feedwater from all three modules
- Trip of intermediate pumps in all three modules

The results show that the plant would respond smoothly to these events, maintaining the reactor at a stable, coolable condition. For the first event, instantaneous loss of IHTS, in which the RVACS is the only heat removal system available, the long-term (~30 hrs) peak core outlet temperature was calculated to compare against design limits. The long-term peak temperature was estimated to be 1058°F on a best-estimate basis (vs. Level B limit of 1100°F), and 1143°F on a conservative basis (vs. Level C limit of 1200°F).

15.5 Comment

Please provide the EM pump characteristic curves used in the analyses in Chapter 5.

Response

See Figure F15.5-1 attached.

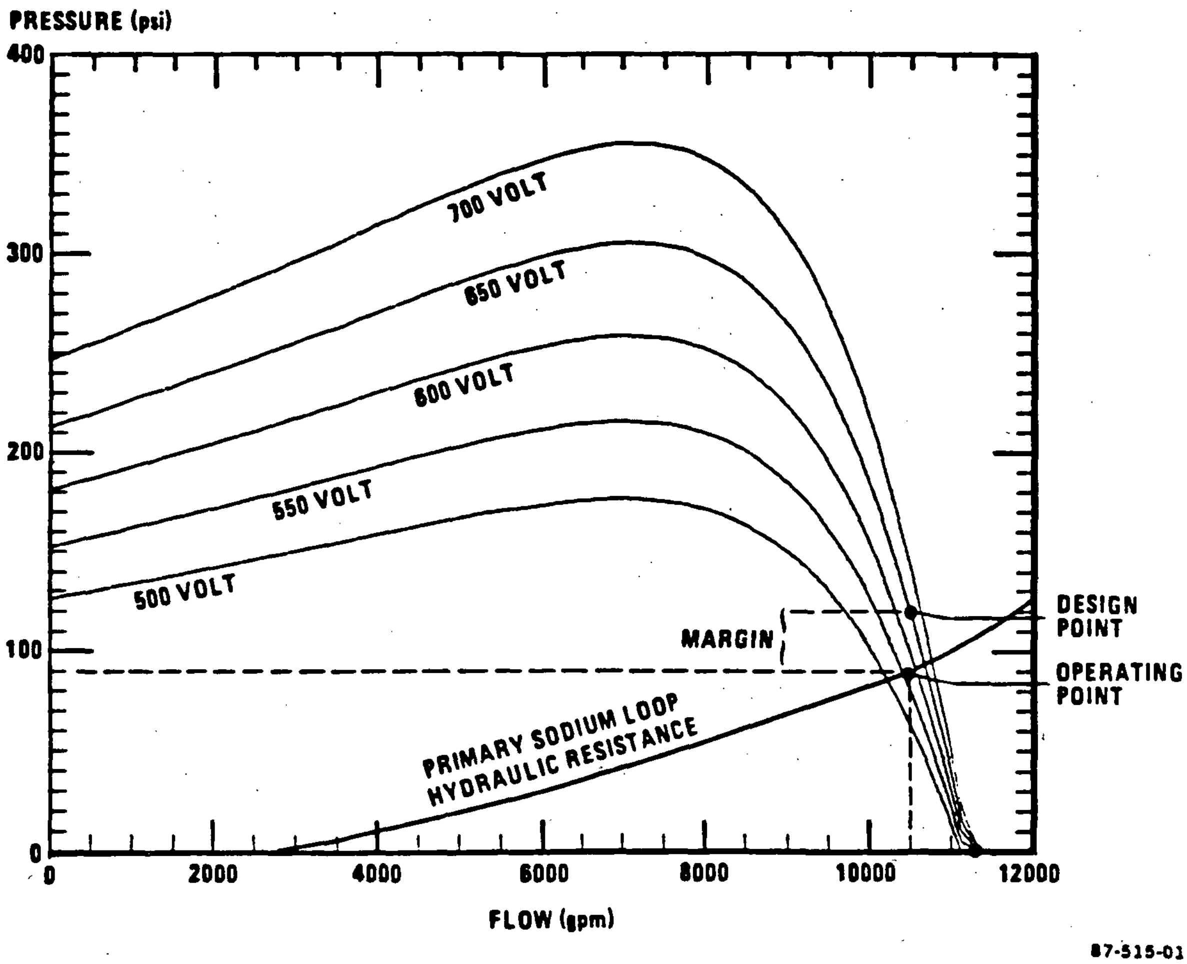


Figure 15.5–1 PRIMARY EM PUMP PERFORMANCE CURVES

15.6 **Comment**

Please provide detailed analysis plots for the first 50-60 seconds for all results reported in the analyses in the PSID.

Response

Detailed plots for the design basis events noted below are attached:

Fast Runback Caused by Single Rod Withdrawal	Figs. F15.6-1a through F15.6-1e
Decay Heat Removal by Auxiliary Cooling System	Figure F15.6-1e
Core Outlet Temperatures With and Without RVACS Fouling	Figure F15.6-3

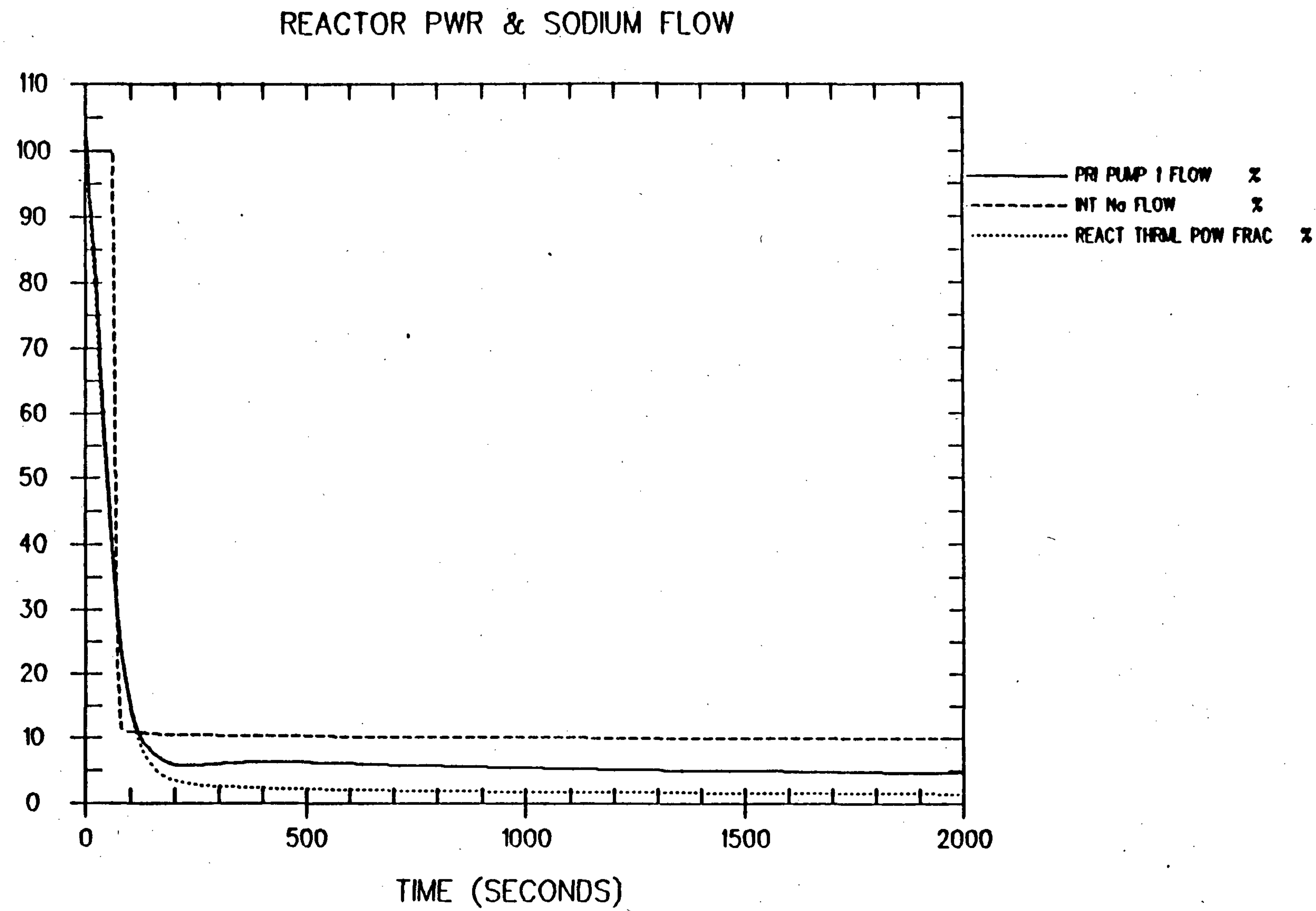


FIGURE 15.6-1a REACTOR POWER AND PHTS AND IHTS FLOWS DURING FAST RUNBACK CAUSED BY SINGLE ROD WITHDRAWAL

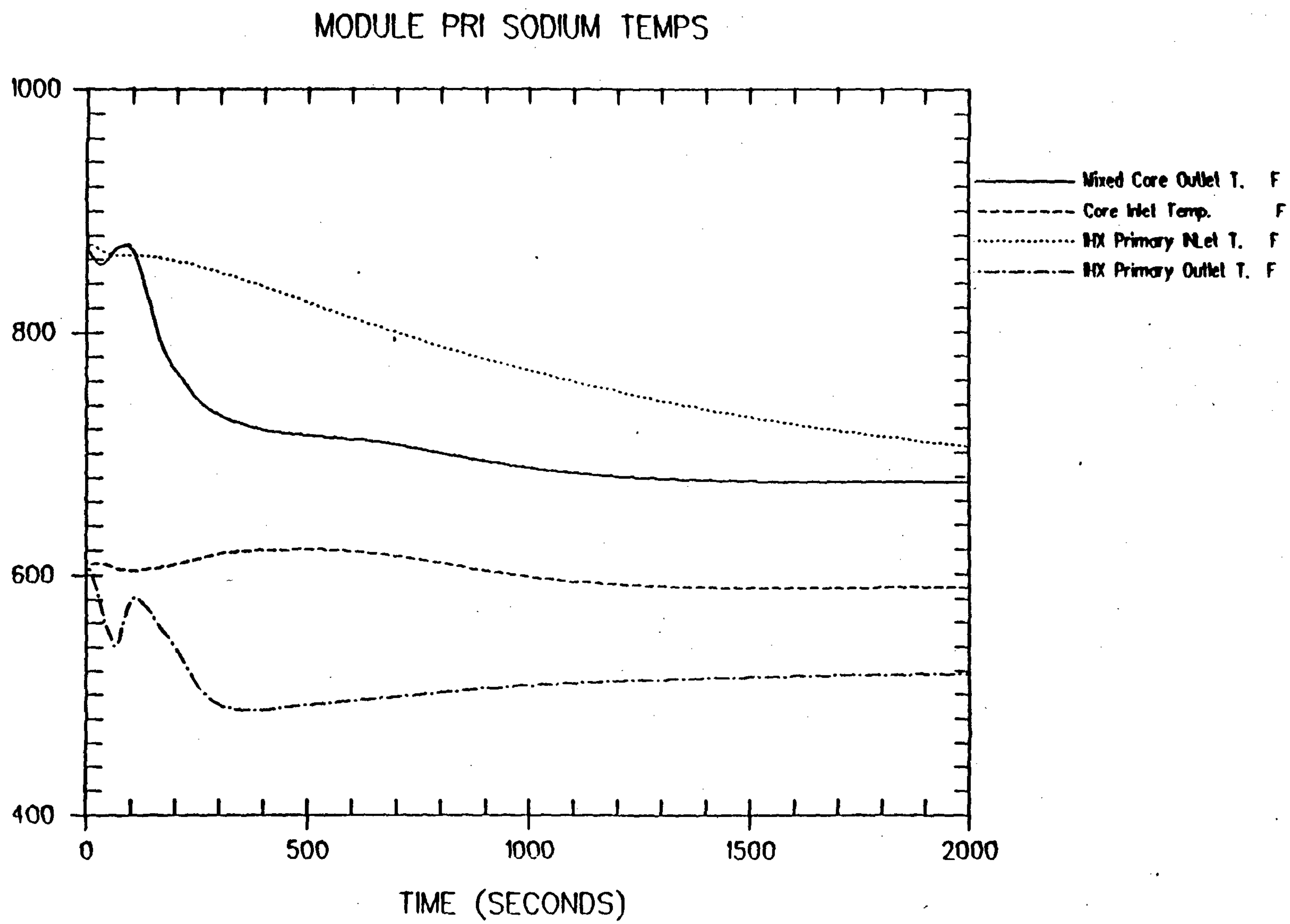


FIGURE 15.6-1b PRIMARY SODIUM TEMPERATURES DURING FAST RUNBACK CAUSED BY SINGLE ROD WITHDRAWAL

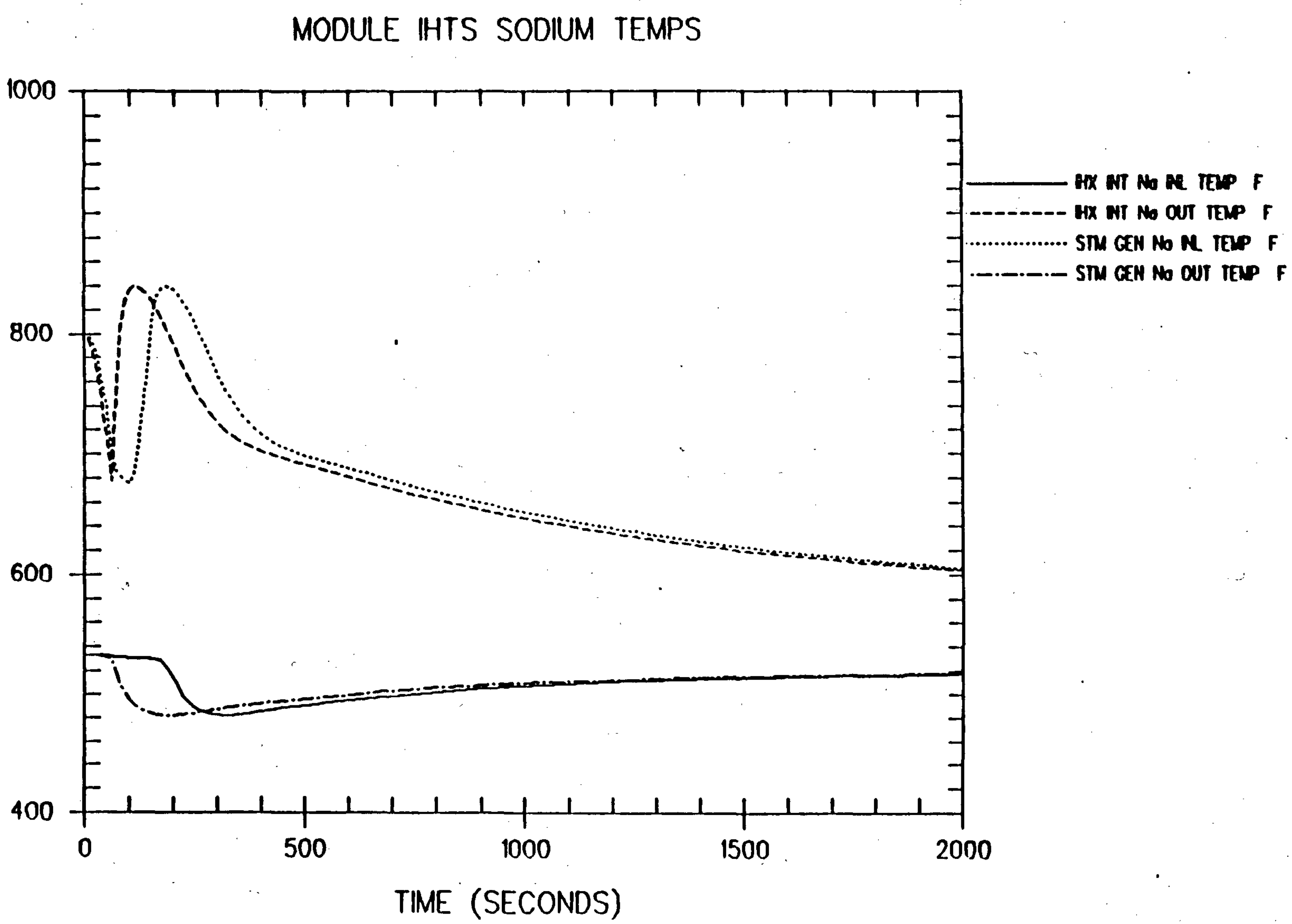


FIGURE 15.6-1c IHTS SODIUM TEMPERATURES DURING FAST RUNBACK CAUSED BY SINGLE ROD WITHDRAWAL

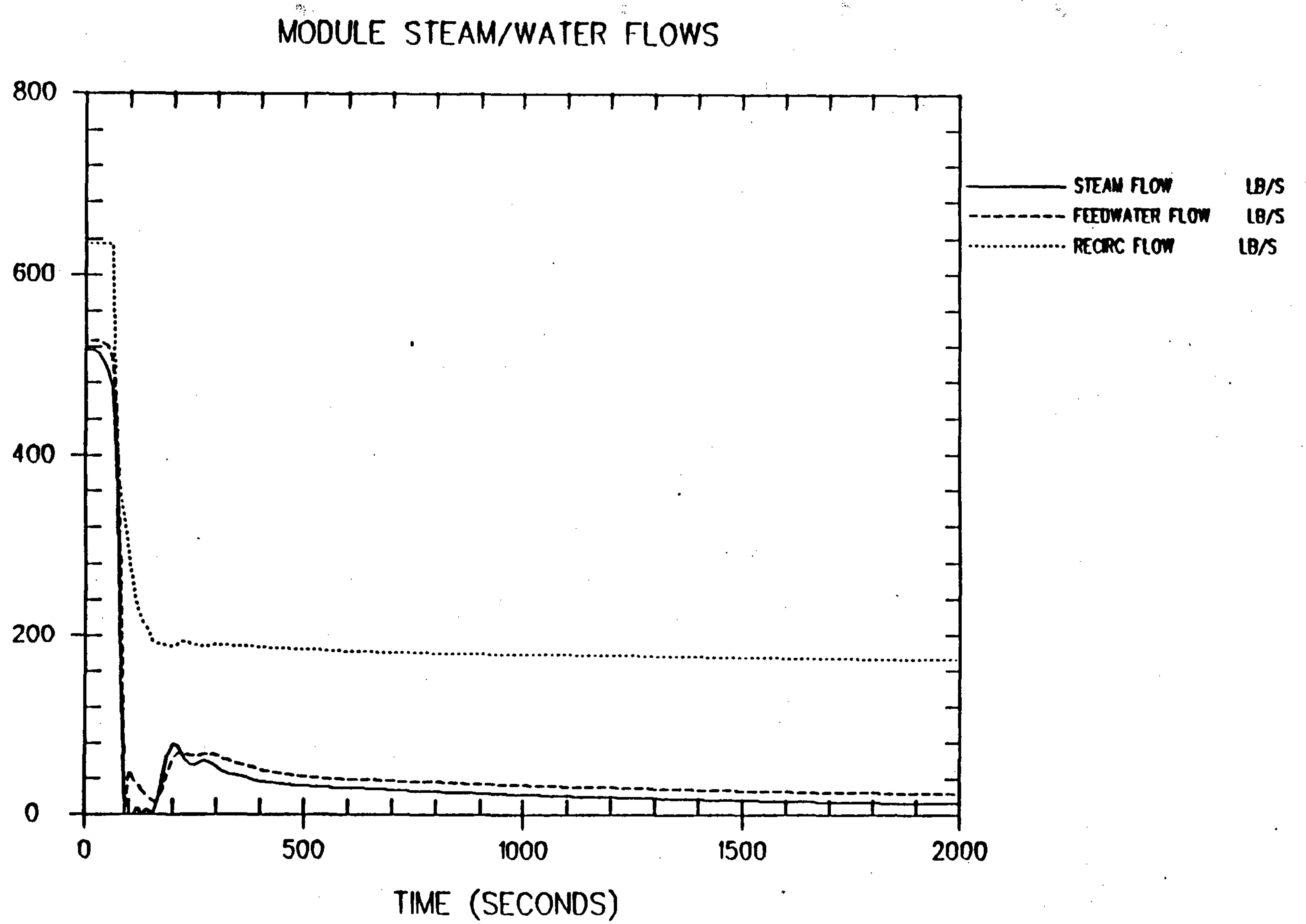


FIGURE 15.6-1d MODULE STEAM AND WATER FLOWS DURING FAST RUNBACK CAUSED BY SINGLE ROD WITHDRAWAL

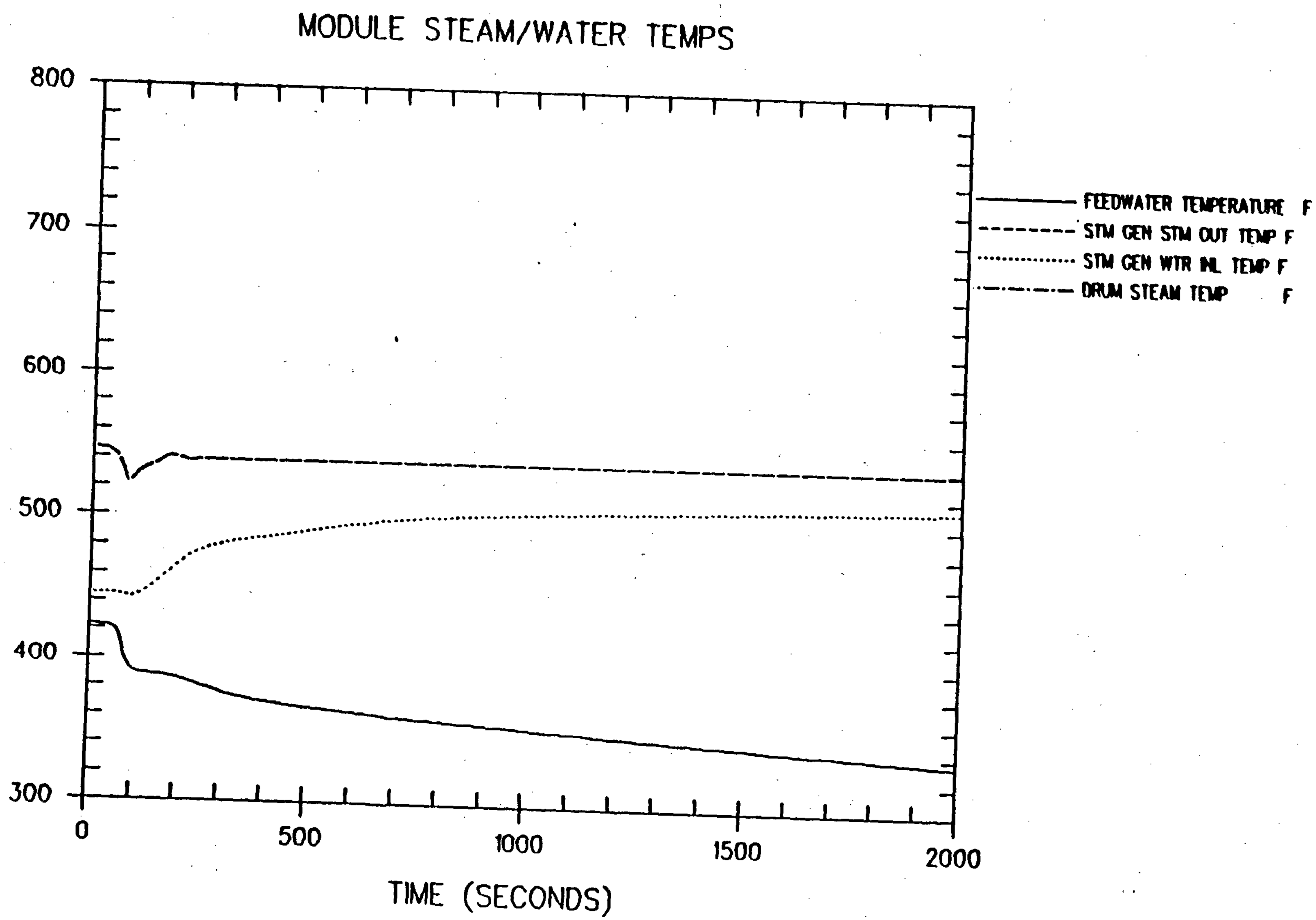


FIGURE 15.6-1e MODULE STEAM AND WATER TEMPERATURES DURING FAST RUNBACK CAUSED BY SINGLE ROD WITHDRAWAL

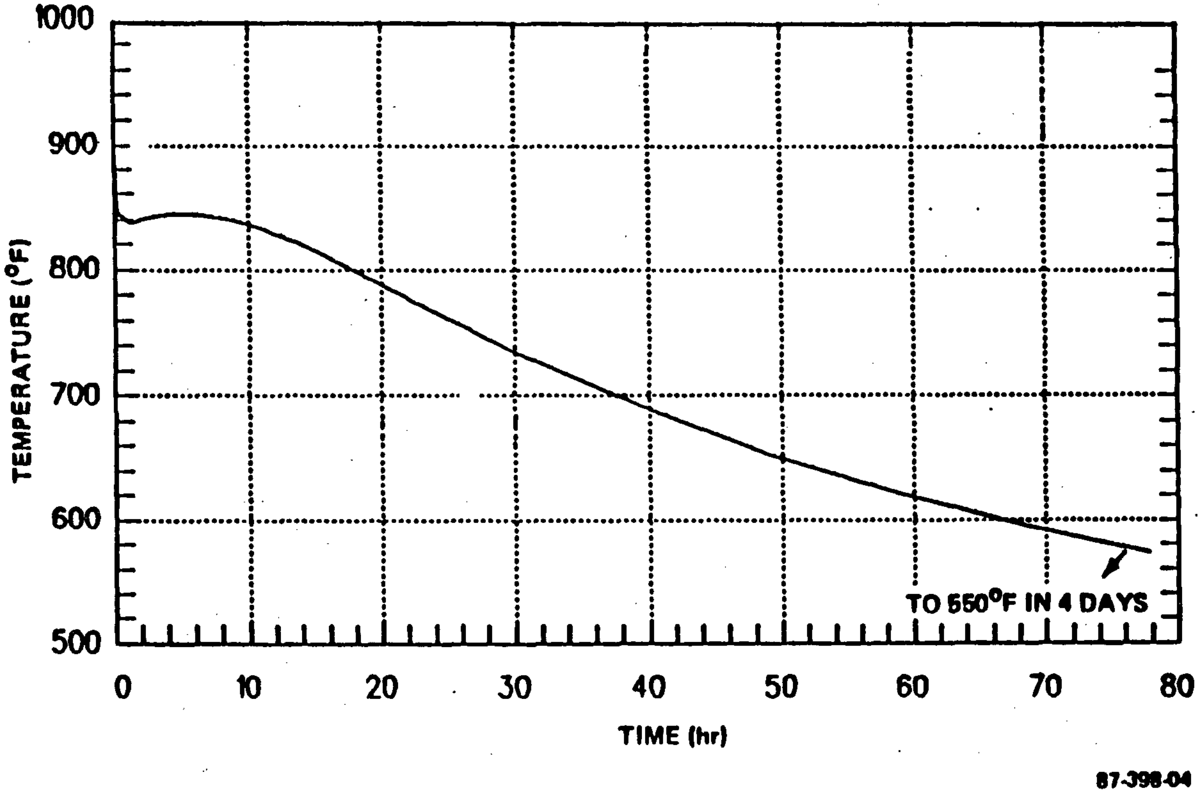


Figure 15.6– 2 DECAY HEAT REMOVAL BY AUXILIARY COOLING SYSTEM (Average Core Outlet Temperature)

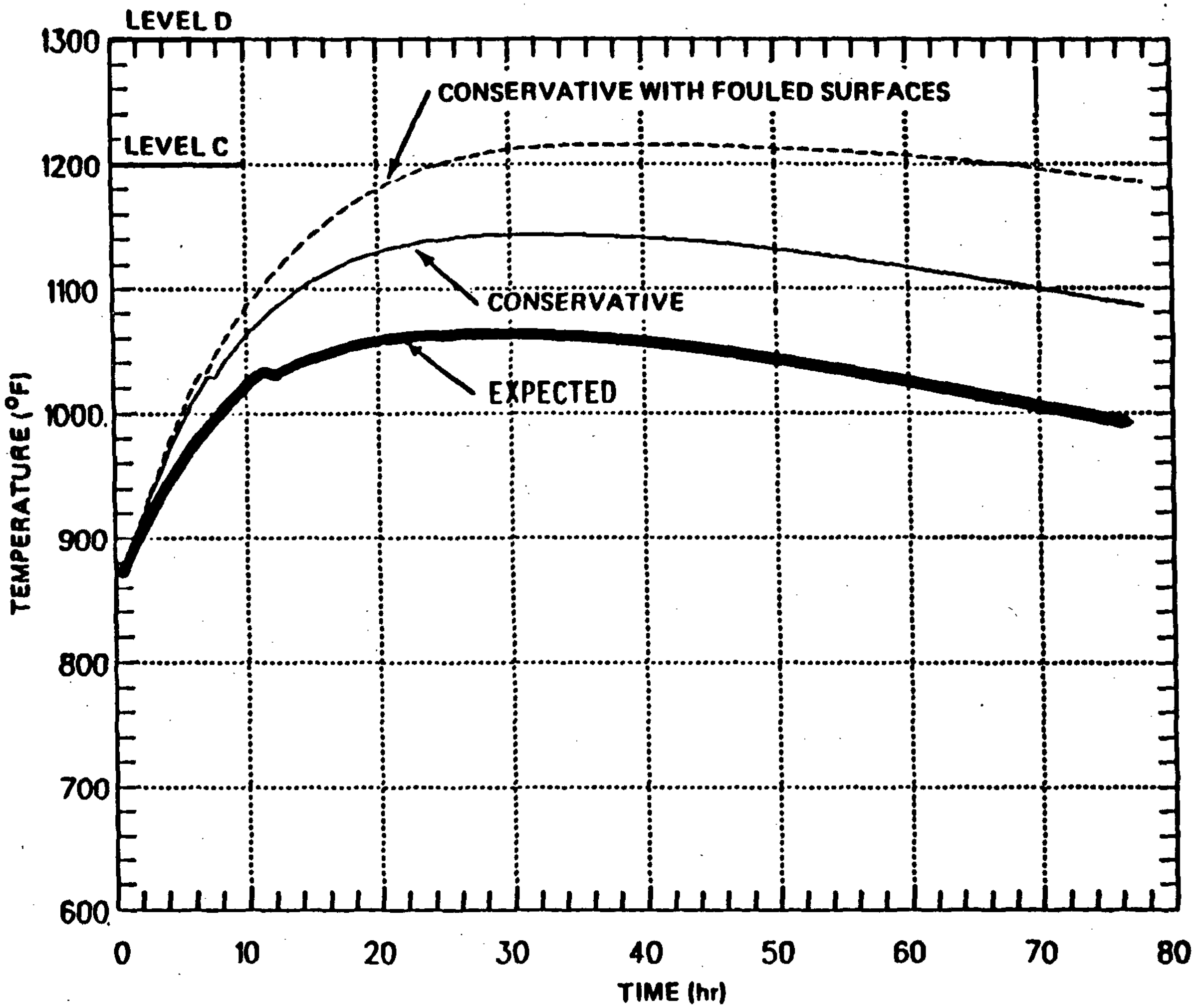


Figure 15.6–3 CORE OUTLET TEMPERATURES WITH AND WITHOUT RVACS FOULING

A

RESPONSES TO NRC COMMENTS ON PSID APPENDIX A

I. GENERAL

AI-1a **Comment**

The use of beta factors in the quantification process is highlighted. A detailed discussion of the common cause analysis and coupled hardware analysis related to the event tree branch point probability within NSSS scope should be provided.

Response

A discussion of the dependencies on the initiating events, between systems, and within systems is included in the updated PRA in Amendment 8 issued November 20, 1948. Please refer to the response to Comment AIII-2 for related responses.

AI-1b **Comment**

Please provide a discussion of the reliability data, including the original source of the data. Many of the data are now presented and used in the PRA without any indication as to their origin.

Response

Data sources used in the PRA are summarized in Table A3.2-1. Direct referencing to the data sources is also included as appropriate for the probabilities of initiating events, system responses and phenomenological scenarios. In general, the data used has been based on the following sources:

1. Reliability data of hardware components not in a sodium environment: the Reactor Safety Study (Wash 1400, NUREG 75/014), Nuclear Plants Reliability Data Systems (NUREG/CR2232), Generating Availability Data System Reports (GADS, issued by NAERC), CRBR PRA (EGG-EA6162).

2. Generic component seismic fragility data: Reliability Data Required for a Seismic Risk Environment (UCRL92798, LLNL).

3. Structural and Sodium Component Reliability Data: Reliability Data for CRBRP SHRS (GEFR-00554), CRBR PRA, Centralized Reliability Data Organization (CREDO), SG Worldwide Tube Performance, Analysis of the 1983-1984 Statistics, (NEI, June 1986), An Assessment of the Integrity of Pressure Vessels (UKAEA, 1982).

4. Monitoring, Testing, Repair Outage Data: The Reactor Safety Study, PRISM systems maintenance, inspection, and outage requirements as defined in the system design documents.

5. Phenomenological Responses and Uncertainties: e.g., thermal hydraulics neutronic analysis, radioactivity transport, accident analysis, structural analysis; probabilistic assessments in these areas are based on experts' judgment based on available computer analysis and bounding analysis. These in turn are based on test data, physical laws, or idealized models.

We appreciate the importance of referencing reliability data bases in a PRA. For the limited PRA scope considered so far, the approach has been to provide local referencing whenever data is used. As the PRA scope expands, a separate section on data will be included in the PRA with listing of the various components, reliability data, uncertainty ranges, and sources.

AI-1c <u>Comment</u>

All results are essentially "best estimate", that is, there does not appear to be any uncertainty analysis or even any discussion regarding the uncertainties associated with the failure probabilities. Discussion of uncertainties should be provided.

<u>Response</u>

In the context of the "Probabilistic Safety Analysis Procedures Guide," (NUREG/CR 2815) the PRISM PRA in Appendix A of the PSID is a "Baseline Evaluation." As correctly characterized by the comment, the evaluation provides "best estimate" results. Although the "RISKSP" code has the capability of propagating uncertainties of various data elements in the risk model, budget and time constraints did not allow exploiting this capability at this time. Nevertheless, and as stated in NUREG/CR2815, the baseline evaluations conducted so far has provided valuable insights in the reliability and safety of the plant.

Future updates of the PRISM PRA are planned to provide insights into the major contributors to risk and significance of uncertainty of data related to these contributors.

AI-1d <u>Comment</u>

For the dominant sequences, please provide a measure of the margins or conservatisms.

<u>Response</u>

Section A2.0 of the PRA identifies the dominant risk constrictors. A preliminary assessment of the risk margin of conservatism has been recently completed. The assessment included the two following tasks:

1. Using the bounding frequencies of initiating events and safety system probabilities provided by the NRC reviewers, re-estimate the public risk for PRISM.

2. Assess the margin in the source term of energetic events.

The bounding frequencies of initiating events and safety system probabilities provided by the NRC reviewers are presented in Tables AI-1 through 3. As seen from these tables, the assumed values are unrealistically conservative and violate the PRISM design requirements. For example: 1) Table AI-1 leads to an initiating event frequency of more than 18 events/yr, 2) Table AI-2 leads to a failure probability of $\geq 10^{-5}$ per demand for the RPS, RSS, and PCDS, and a failure probability of $\geq 10^{-4}$ per demand for RVACS. The above tables were used in the PRISM PRA. Table AI-4 presents the resulting frequency, individual

Table AI-1

INITIATING EVENTS FREQUENCY ASSUMED BY NRC REVIEWERS (PER YEAR)

	Initiating Event	Frequency
1.	Reactivity Insertion 0.07 - 0.18$	10^{-2}
2.	Reactivity Insertion 0.18 - 0.36$	10^{-2}
12.	Loss of Operating Power Heat Removal (Failure of Main Feedwater Control Valve)	1.8
16.	Station Blackout	10^{-1}
17.	Large Na-H_2O Reaction	3×10^{-6}
18.	Spurious Scram and Transients Inadequately Handled by PCS	10
	All Other Initiating Events	Same as PSID, Appendix A

Table AI-2

SYSTEM EVENT TREES PROBABILITIES ASSUMED BY NRC REVIEWERS (PER DEMAND)

System		Conditional Failure Probability
RPS		10^{-5} For IE Other Than IE6 (Earthquakes >0.825g)
		10^{-3} for IE6
RSS		10^{-5} For IE Other Than IE6 or Vessel Rupture
		3×10^{-2} for IE6
		1 For Vessel Rupture
PCDS		10^{-5} for IE Other Than IE6
		5.83×10^{-1} for IE6
SHRS:	ACS:	4×10^{-3} For IE's Other Than The Following:
		IE4: 6×10^{-3}
		IE5: 1×10^{-3}
		IE6, 7, 14, 17: 1
	RVACS:	10^{-4} For IE's Other Than RVACS Blockage or IE6 With Isolators Failure
		I Otherwise

Table AI-3

NRC ASSUMED EUTECTIC AND Na BOILING PROBABILITIES

GIVEN UNPROTECTED INITIATING EVENTS

Initiating Event	Probability Given Unprotected IE Eutectic	 Na-Boiling
1. Reactivity Insertion 0.07-0.18$	0.01	0
2. Reactivity Insertion 0.18-0.36$	0.05	0.01
3. Reactivity Insertion >0.36$	0.5	0.1
4. Earthquake 0.3g-0.375g	0.01	0
5. Earthquake 0.375g-0.825g	0.05	0.01
6. Earthquake >0.825g	0.5	0.1
7. Vessel Fracture	0.5	0.1
8. Local Coolant Blockage	0.01	0
9. Vessel Leak	0.01	0
10-18	0.05	0.01

Table AI-4 - ESTIMATED ACCIDENT FREQUENCY AND RISKS FROM NRC DATA

Accident		Type	Frequency	Individual Risk*	Societal Risk*
1.	S3	LOSHR	8(-6)	8(-11)	2(-9)
2.	S5	LOSHR	7(-11)	7(-16)	2(-14)
3.	P1	UTOP	1(-7)	3(-19)	6(-18)
4.	P2	UTOP	1(-7)	2(-16)	9(-18)
5.	P3	UTOP	1(-11)	3(-15)	1(-16)
6.	P4	UTOP	1(-17)	8(-20)	4(-21)
7.	P1S	UTOP/LOSHR	2(-17)	2(-22)	5(-21)
8.	P2S	UTOP/LOSHR	2(-17)	2(-22)	4(-21)
9.	P3S	UTOP/LOSHR	2(-20)	6(-24)	7(-23)
10.	P4S	UTOP/LOSHR	8(-27)	7(-29)	4(-30)
11.	F1	ULOF	1(-4)	3(-13)	1(-14)
12.	F3	ULOF	1(-9)	6(-12)	3(-13)
13.	F3S	ULOF/LOSHR	6(-16)	3(-18)	1(-19)
14.	H2	ULOHS	1(-5)	1(-13)	3(-15)
15.	H3	ULOHS	1(-10)	5(-13)	2(-14)
16.	H1S	ULOHS/LOSHR	2(-11)	2(-16)	5(-15)
17.	H2S	ULOHS/LOSHR	6(-12)	3(-16)	2(-15)
18.	H3S	ULOHS/LOSHR	2(-17)	8(-20)	6(-21)
19.	G3	UTOP/ULOF	9(-9)	4(-12)	2(-13)
20.	G4	UTOP/ULOF	1(-8)	2(-10)	6(-12)
21.	G1S	UTOP/ULOF/LOSHR	5(-14)	6(-19)	1(-17)
22.	G3S	UTOP/ULOF/LOSHR	9(-13)	4(-16)	2(-16)
23.	G4S	UTOP/ULOF/LOSHR	1(-9)	1(-11)	5(-13)
	TOTAL			3(-10)	2(-9)
	GOAL			5(-7)	2(-6)

* With No Evacuation

risk and public risk for the accident types in the PRISM PRA model. The corresponding total individual risk, public risk, and risk goals are also shown at the bottom of the table.

From these results, it is clear that, even with the extremely conservative frequencies and probabilities assumed, the PRISM individual risk and public risk are bout three orders of magnitude lower than the safety goals.

An assessment of the source term energetic scenarios in the innovative design PRA's has been recently completed by ANL. The assessment recommends use of 1% fuel inventory vapor source term as a bound for release in energetic scenarios. The rationale for using this bounding source term is presented in the response to Comment AXII-ANL-2. This bounding source term, compared to the 10% fuel source term conservatively assigned to the PRISM high energetic scenarios, shows the high energetic fuel release fraction used in the PRISM PRA is conservative by a factor of 10.

AI-1e <u>Comment</u>

A discussion of the sequences in the event trees is needed to explain why some top events are "don't care" events in some sequences and why some top events have zero or one probability in some sequences.

<u>Response</u>

A discussion of the sequences in the event tree, and event dependencies which cover the above specific cases is included in Amendment 8 issued November 20, 1987.

AI-1f <u>Comment</u>

An additional discussion of the success criteria for the purpose of defining a core melt following an initiating event should be provided. The discussion could include any potential sources of uncertainty contribution.

<u>Response</u>

This discussion is provided in the PRA update in Amendment 8 issued November 20, 1987.

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AI-2a Comment

A separate discussion of the major dominant core melt sequences, dominant contributors to these sequences and contributors to public risk should be provided. Such a discussion will be very useful to understand the strength and weaknesses of the proposed design. As part of identification of the dominant contributors to core melt and risk, a sensitivity and/or importance measure calculations could be performed and provided to the staff for review.

Response

Dominant sequences and major contributors to risk are identified and discussed in the updated PRA in Amendment 8 issued November 20, 1987.

AI-2b Comment

Also, an "importance analysis" could be used to determine where the maximum benefit could be achieved by plant improvement. In addition, the response of PRISM to the attached list of "PRISM Deterministic BDBA's" should be provided for information.

Response

Importance analyses will be considered as the safety analyses become well defined and reference design features are adequately identified.

The NRC staff has indicated the bounding event sequences list is changing and our response should be deferred at this time.

PRISM-Deterministic BDBA's

These events are intended to bound the PRISM DBA and BDBA spectrum to account for uncertainties and provide for conservatisms in selecting a SSST and assessing the adequacy of containment and off-site evacuation plans. They provide a bounding accident for the following key event categories:

- Reactivity insertion events
- Heat removal events
- Loss of coolant events
- Na/H_2O reaction events

BDBA-1 Inadvertent withdrawal of all control rods-w/o scram for 36 hours (one module):

- o w/forced cooling (primary and secondary pumps continue at 100% flow)
- o w/RVACS cooling only

BDBA-2 36 hour station blackout (all modules) w/failure to scram (one module)

BDBA-3 Loss of forced cooling plus RVACS for 36 hours (one module)

- o with scram
- o without scram

BDBA-4 Instantaneous stoppage of power to one primary pump (one module)

- o with scram
- o without scram

BDBA-5 S.G. tube rupture (all tubes) with failure to isolate or dump S.G. (one module)

BDBA-6 Large Na leaks (one module)

- o Double ended guillotine break of IHTS pipe
- o DEG break of primary pipe with failure to scram
- o RV leak

BDBA-7 External events consistent with their treatment for LWR's.

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AI-3 Comment

A discussion related to the development of the Reliability Assurance Program to update, maintain and use the plant reliability and risk models could be helpful. This could be performed to a level and scope suggested currently by the DOE as part of the MHTGR design review.

Response

The PRISM project implements a Reliability Assurance Program which has been tailored to the needs and scope of the conceptual phase of the project. Some of the primary tasks of this program include:

1. Development of reliability requirements and guidelines.
2. Conducting reliability assessments in support of trade-off studies.
3. Providing reliability assessments in support of PRA.
4. Establishment of a reliability data base.
5. Participating in design reviews.

The program has provided input for such decisions as reliability and availability allocation to systems, subsystems and components, selection of reactor shutdown system concept, and the selection of a steam generator concept. The program has been effective for the total project scope. The need of operation involving close interaction with the designer has been particularly productive in identifying potential vulnerabilities and strengths of design options. As the PRISM design moves to its next phase of design, a more extensive reliability program will be instituted.

AI-4 Comment

There is some difficulty in putting the results of the PRA in perspective due to the lack of sufficient intermediate probability results. Please provide tables of:

(a) Accident type frequencies.

(b) Plant damage state frequencies for each accident type.

(c) "Total" plant damage state frequencies.

(d) Release category frequencies for each plant damage state.

(e) "Total" release category frequencies.

Response

The requested frequencies are attached. Please see Appendix A of the PSID for definition of the accident types, damage categories, and release categories.

~~D-MA~~

FREQUENCIES OF CORE DAMAGE CATEGORIES

	FREQ	C1 ~~CD-1~~	C2 ~~CD-2~~	C3 ~~CD-3~~	C4 ~~CD-4~~	C5 ~~CD-5~~	C6 ~~CD-6~~	C1S ~~CD-7~~	C2S ~~CD-8~~	C3S ~~CD-9~~
S3 ~~AT-1~~	5.11E-12 ~~4.00E-08~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	1.00E+00 (100.0000)	1.00E-37 (0.0000)	0.00E+00 (0.0000)
S5 ~~AT-2~~	3.0 E-11 ~~2.44E-12~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	1.00E+00 (99.9698)	1.00E-37 (0.0000)
P1 ~~AT-3~~	2.8 E-13 ~~1.22E-13~~	9.90E-01 (0.0007)	1.01E-02 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P2 ~~AT-4~~	9.64E-12 ~~1.22E-13~~	9.50E-01 (0.0007)	4.48E-02 (0.0001)	4.93E-03 ~~(0.0448)~~	5.43E-05 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P3 ~~AT-5~~	1.44E-15 ~~1.21E-14~~	5.04E-01 (0.0000)	2.46E-01 (0.0001)	1.23E-01 ~~(0.1109)~~	6.63E-02 (0.0002)	3.08E-02 (0.0001)	3.08E-02 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P4 ~~AT-6~~	1.3 E-21 ~~1.24E-20~~	1.22E-02 (0.0000)	9.47E-02 (0.0000)	4.55E-03 (0.0000)	8.53E-02 (0.0000)	8.03E-02 (0.0000)	7.23E-01 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P1S ~~AT-7~~	1.0 E-28 ~~2.55E-24~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.90E-01 (0.0000)	1.01E-02 (0.0000)	0.00E+00 (0.0000)
P2S ~~AT-8~~	6.24E(-29) ~~2.56E-24~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.50E-01 (0.0000)	4.48E-02 (0.0000)	4.93E-03 (0.0000)
P3S ~~AT-9~~	5.0 E-24 ~~1.13E-24~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	5.04E-01 (0.0000)	2.46E-01 (0.0000)	1.23E-01 (0.0000)
P4S ~~AT10~~	2.3 E-33 ~~2.86E-27~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	1.22E-02 (0.0000)	4.89E-01 (0.0000)	2.51E-03 (0.0000)
F1 ~~AT11~~	5.5 E-8 ~~1.70E-08~~	9.90E-01 (99.9892)	1.01E-02 ~~(3.0416)~~ 14.385	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
F3 ~~AT12~~	6.6 E-9 ~~6.89E-09~~	0.00E+00 (0.0000)	4.94E-01 ~~(95.0891)~~ 84.342	0.00E+00 (0.0000)	5.35E-02 ~~(96.3695)~~ 47.63	4.58E-02 ~~(97.8646)~~ 55.67	4.51E-01 ~~(80.9810)~~ 12.62	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
F3S ~~AT13~~	6.0 E-21 ~~6.94E-17~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	4.94E-01 (0.0014)	0.00E+00 (0.0000)
H2 ~~AT14~~	8.3 E-11 ~~4.98E-17~~	9.90E-01 ~~(0.0000)~~ 0.015	1.01E-02 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
H3 ~~AT15~~	7.0 E-11 ~~6.10E-17~~	0.00E+00 (0.0000)	4.49E-01 (0.0000)	0.00E+00 (0.0000)	7.39E-02 ~~(0.0000)~~ 0.698	5.82E-02 ~~(0.0000)~~ 0.75	4.19E-01 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
H1S ~~AT16~~	2.9 E-23 ~~7.75E-20~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.90E-01 (0.0000)	1.01E-02 (0.0000)	0.00E+00 (0.0000)
H2S ~~AT17~~	2.5 E-18 ~~4.83E-24~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.90E-01 (0.0000)	1.01E-02 (0.0000)	0.00E+00 (0.0000)
H3S ~~AT18~~	3.55E-29 ~~6.20E-25~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	4.52E-01 (0.0000)	4.48E-02 (0.0000)
G8 ~~AT19~~	5.9 E-11 ~~5.53E-12~~	3.02E-01 ~~(0.0094)~~	3.14E-01 ~~(0.0370)~~	1.57E-01 ~~(64.6568)~~	1.42E-01 ~~(0.1595)~~	4.51E-02 ~~(0.0597)~~	3.99E-02 (0.0045)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)

G4 ~~AT20~~	2.13E-8 ~~9.74E-10~~	0.00E+00 (0.0000)	1.47E-03 (~~0.0312~~) 0.81	4.84E-04 (~~35.1876~~) 53.53	1.76E-02 (~~3.4708~~) 50.58	1.10E-02 (~~2.5755~~) 43.15	9.69E-01 (~~10.0637~~) 87.38	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
G1S ~~AT21~~	5.0E-23 ~~2.62E-15~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.90E-01 (0.0000)	1.01E-02 (0.0011)	0.00E+00 (0.0000)
G3S ~~AT22~~	2.5E-18 ~~1.68E-15~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	3.02E-01 (0.0000)	3.14E-01 (0.0216)	1.57E-01 (84.2592)
G4S ~~AT23~~	9.59E-10 ~~1.02E-13~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	1.47E-03 (0.0061)	4.84E-04 (~~15.7408~~) 100
TOTA		~~1.77E-08~~ 5.46(-8)	~~4.58E-09~~ 3.86E-9	~~1.84E-12~~ 1.9254(-11)	~~4.93E-10~~ 7.41E-10	~~4.18E-10~~ 5.43E-10	~~4.05E-09~~ 2.362(-8)	~~4.08E-08~~ 5.11E-12	~~2.45E-12~~ 3.0E-11	~~8.10E-10~~ 4.64(-13)

	FREQ	CD10	CD11	CD12
S3 ~~AT 1~~	5.11E-12 ~~4.03E-08~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
S5 ~~AT 2~~	3.0E-11 ~~2.44E-12~~	1.00E-37 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P1 ~~AT 3~~	2.8E-13 ~~1.22E-13~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P2 ~~AT 4~~	9.64E-12 ~~1.12E-13~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P3 ~~AT 5~~	1.44E-15 ~~1.21E-14~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P4 ~~AT 6~~	1.3E-21 ~~1.24E-20~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P1S ~~AT 7~~	4.0E-28 ~~2.55E-24~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P2S ~~AT 8~~	6.24E-29 ~~2.50E-24~~	5.43E-05 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
P3S ~~AT 9~~	5.0E-24 ~~1.13E-24~~	6.63E-02 (0.0000)	3.08E-02 (0.0000)	3.08E-02 (0.0000)
P4S ~~AT10~~	2.3E-33 ~~2.06E-27~~	4.72E-02 (0.0000)	4.44E-02 (0.0000)	4.04E-01 (0.0000)
F1 ~~AT11~~	5.5E-8 ~~1.79E-08~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
F3 ~~AT12~~	6.6E-9 ~~8.89E-08~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
F3S ~~AT13~~	6.5E-21 ~~6.04E-17~~	5.35E-02 (0.1827)	4.58E-02 (0.2640)	4.51E-01 (0.0317)
H2 ~~AT14~~	8.3E-11 ~~4.08E-17~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
H1S ~~AT15~~	7.0E-11 ~~6.19E-17~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
H2S ~~AT16~~	2.9E-23 ~~7.75E-20~~	0.00E+00	0.00E+00	0.00E+00

				(0.0000)	(0.0000)	(0.0000)
H3S	~~AT17~~	~~4.03E-24~~	2.5E-18	0.00E+00	0.00E+00	0.00E+00
				(0.0000)	(0.0000)	(0.0000)
63	~~AT18~~	~~8.20E-25~~	3.55E-29	5.08E-02	4.67E-02	4.06E-01
				(0.0000)	(0.0000)	(0.0000)
G4	~~AT19~~	~~5.53E-12~~	5.7E-11	0.00E+00	0.00E+00	0.00E+00
				(0.0000)	(0.0000)	(0.0000)
G1S	~~AT20~~	~~8.74E-16~~	2.13E-8	0.00E+00	0.00E+00	0.00E+00
				(0.0000)	(0.0000)	(0.0000)
G3S	~~AT21~~	~~2.62E-15~~	5.0 E-23	0.00E+00	0.00E+00	0.00E+00
				(0.0000)	(0.0000)	(0.0000)
G4S	~~AT22~~	~~1.68E-15~~	2.5E-18	1.42E-01	4.51E-02	3.99E-02
				(11.7868)	(6.3135)	(0.0681)
	AT23	~~1.02E-13~~	9.59E-10	1.76E-02	1.10E-02	9.69E-01
				~~(88.0305)~~ 100.0	~~(93.4225)~~ 100.0	~~(99.9002)~~ 100.0
	TOTA			~~2.03E-15~~ 1.69E-11	~~1.20E-15~~ 1.05E-11	~~9.87E-14~~ 9.29E-10

~~G MA~~

FREQUENCIES OF RELEASE CATEGORIES

	FREQ	R2A ~~RC 1~~	R3 ~~RC 2~~	P4A ~~RC 3~~	P6A ~~RC 4~~	R6U ~~RC 5~~	R6S ~~RC 6~~	R8A ~~RC 7~~	R8U ~~RC 8~~	R8S ~~RC 9~~
C1	~~CD 1 1.77E-08~~ 5.46E-8	9.95E-38 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.40E-07 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.41E-09 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
C2	~~CD 2 4.50E-09~~ 3.86E-9	9.95E-38 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.40E-07 ~~(0.0454)~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.41E-09 ~~(0.0454)~~	0.00E+00 (0.0000)
C3	~~CD 3 1.04E-12~~ 1.93E-11	9.95E-38 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.40E-07 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.41E-09 (0.0000)	0.00E+00 (0.0000)
C4	~~CD 4 4.93E-10~~ 7.41E-10	9.96E-05 ~~(1.1718)~~ 1.32	0.00E+00 (0.0000)	9.97E-07 (0.0000)	0.00E+00 (0.0000)	2.40E-07 ~~(0.0040)~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.41E-09 ~~(0.0040)~~	0.00E+00 (0.0000)
C5	~~CD 5 4.10E-10~~ 5.43E-10	9.92E-03 ~~(98.8284)~~ 96.77	0.00E+00 (0.0000)	9.93E-05 ~~(0.0009)~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.38E-07 ~~(7.7799)~~	0.00E+00 (0.0000)	0.00E+00 (0.0000)	2.38E-09 ~~(7.7798)~~
C6	~~CD 6 4.95E-08~~ 2.36E-8	0.00E+00 (0.0000)	9.33E-02 ~~(99.9980)~~ 96.2	9.07E-01 ~~(99.9971)~~ 96.2	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
C1S	~~CD 7 4.03E-08~~ 5.11E-12	9.95E-38 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.90E-01 (100.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.91E-03 (100.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
C2S	~~CD 8 2.45E-12~~ 3.0E-11	9.95E-38 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.90E-01 ~~(99.8539)~~ 63.38	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.91E-03 ~~(99.8539)~~ 63.34	0.00E+00 (0.0000)
C3S	~~CD 9 3.13E-16~~ 4.64E-13	9.95E-38 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.90E-01 ~~(0.0128)~~ 1.0	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.91E-03 ~~(0.0128)~~ 1.0	0.00E+00 (0.0000)
C4S	~~CD10 2.03E-15~~ 1.69E-11	9.96E-05 (0.0000)	0.00E+00 (0.0000)	9.97E-07 (0.0000)	0.00E+00 (0.0000)	9.90E-01 ~~(0.0830)~~ 35.64	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.91E-03 ~~(0.0830)~~ 35.64	0.00E+00 (0.0000)
C5S	~~CD11 1.20E-15~~ 1.05E-11	9.92E-03 ~~(0.0003)~~ 1.87	0.00E+00 (0.0000)	9.93E-05 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.80E-01 ~~(92.2201)~~ 100.0	0.00E+00 (0.0000)	0.00E+00 (0.0000)	9.81E-03 ~~(92.2201)~~ 100.0
C6S	~~CD12 9.87E-14~~ 9.29E-10	0.00E+00 (0.0000)	9.33E-02 ~~(0.0020)~~ 3.8	9.07E-01 ~~(0.0020)~~ 3.8	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)	0.00E+00 (0.0000)
	TOTA	~~4.20E-12~~ 5.57E-12	~~4.62E-10~~ 2.29E-9	~~4.49E-09~~ 2.22(-8)	~~3.99E-08~~ 5.06 (-13)	~~2.42E-12~~ 4.69(-11)	~~1.20E-15~~ 1.03E-11	~~3.99E-10~~ 5.06 E-14	~~2.40E-14~~ 4.69(-13)	~~1.20E-17~~ 1.03E-13

AI-5 Comment

A bounding type qualitative and/or quantitative analysis of external events, other than seismic events, consistent with the scope of the conceptual design should be provided. The guidelines provided in the NUREG 2815 could be adopted to perform the external events analysis.

Response

Design basis external events (DBEE) have been defined for PRISM in Chapter 3 of the PSID. These DBEE have been selected on the basis of past experience. Exceeding these DBEE's in the conceivable sites for PRISM is expected to be extremely unlikely, since the selection of these sites will be based on showing that the probability of exceeding the DBEE's is extremely small. Protection against these DBEE's is provided.

PRISM conceptual design effort does not allow the development of hazard curves and fragilities for external events other than seismic events, as suggested by NUREG 2815. As generic hazard curves for the USA regions become available from other programs, (i.e., Draft NUREG 4812 - Contents of PRA Submittal for LWR's), the project will reconsider the susceptibility of PRISM to external events. Evaluations will confirm that the BDBE's are adequately bounded to accept PRISM siting.

AI-6 Comment

Given an initiating event, how can the success of a system result in continued safe operation of the plant (page A4-1 second paragraph)?

Response

The cited paragraph states that:

> "Section A4.2 displays the possible responses of the module systems to each initiating event. Systems of interest include those designed to control the module power, coolant flow, and heat removal. The possible success and failure modes of these systems may lead to safe shutdown, continued safe operation, or one of twenty-three accident types."

The PRISM design has inherent capabilities to override some initiating events without challenging the safety limits of the fuel, clad, or coolant, even under the hypothetical assumption that the reactor shutdown system fails to scram in response to the initiating event. For example, an unprotected transient overpower initiated by accidental full withdrawal of a control rod without scram leads to a power increase which stabilizes at 103% of nominal power. If coolant flow as heat removal capabilities are retained, then the reactor may continue operation virtually indefinitely. The increase of only 3% in the power level is well within the margin of the heat removal system. The plant control system is capable of accommodating such an increase by reducing the power level of other modules in the same power block.

RESPONSES TO NRC COMMENTS ON PSID APPENDIX A

AI-7 Comment

Worldwide reactor trip system experience had unavailability at 1-6 E-5. How does PRISM justify E-9 to E-11? CCF does not appear on the fault trees (Fig 4.2-26 to Fig 4.2-28), why not?

Response

Derivation of the RPS unavailability is presented in Section A4.2 of the PSID. It is interesting to note that component failure data experienced in LWRs have been used to develop the RPS unavailability estimates. The small unavailability estimates obtained reflect the following merits of the PRISM RPS design and operation which are discussed in Section A4.2

1) The system is quadruply redundant with each channel serving sequentially as a standby while the other three channels are performing their reactor protection function. Trip signals from two out of the three channels will trip the reactor.

2) RPS component and channels, including standby ones, are almost continuously monitored.

3) The system has a diagnostic logic with capability to identify, isolate, and reconfigure around a fault so that maintenance and repair in a channel of the faulty equipment can be performed without affecting the functional capability of the remainder of the channel.

4) The system is modularized for high maintainability.

The unavailability models of Section A4.2 include these factors which collectively have resulted in the high performance predicted for PRISM.

The question of common cause failure has not been addressed in the RPS unavailability models due to the lack of adequate design details which will allow adequate evaluation of failure dependencies. The RPS unavailability estimate has been updated recently to include estimates of dependent failures and reflect recent design changes. The updated estimates are included in the Amendment 8 issued November 20, 1987.

AI-8 Comment

Please submit calculations with regard to

a) CCF of control rods
b) CCF of coastdown system
c) CCF of RPS

Response

The CCF evaluations of the reactor shutdown system (RSS) and pump coastdown system are contained in Section A4.2.3 and A4.2.5, respectively. The evaluations include assessment of failure dependence on seismic events and general dependencies expressed using the beta factor. For the RPS, dependence on the initiating events has been evaluated in Section A4.2.2. General dependency evaluation using the beta factor has been conducted for the RPS and is included in Amendment 8 issued November 20, 1987.

It should be noted that the PRISM design is at a conceptual design stage with many of the design and operation details yet to be defined. Consequently, dependence of system responses on the initiating event and functional dependence between systems have been evaluated to the extent possible in Section A4.2. The beta factor approach has been chosen to model dependencies within a system due to the noted conceptual nature of the design.

AI-9 Comment

Please explain the conditional probabilities for IE 5 & 6 in Table A4.2-5.

Response

The conditional probability for any initiating event in Table A4.2-5 is the probability that one or more control rods fail to receive a trip signal from the PCS or RPS given that the initiating event has occurred. As stated in Section A4.2, seismic fragility assessments for the RPS and PCS were made for the SSE only (IE4). For initiating events IE5 and IE6, which present stronger earthquakes, the failure probabilities were increased subjectively to reflect the increased vulnerability to the increase in seismic loading. Later investigation of generic component seismic fragility characteristics indicated that the subjectivity assigned relative increase in the probabilities for IE5 and IE6 may have been too optimistic. These probabilities have subsequently been revised to accommodate recent changes in the PRS and PCS. The revised probabilities are included in Amendment 8 issued November 20, 1987.

AI-10 Comment

Please explain the use of mean time to repair in the PRA, e.g., t_m in Table A4.1-1. How do you use the probabilities in the "p" column of Table A4.1-1? How do you use the failure rates and repair rates in Table A4.2-14? What is the effect of the initiating events on the assessment of the probabilities in Table A4.2-15? What is the mission time for the IHTS pump (block 137)?

Response

The parameters "t_m" in Table A4.1-1 is defined as the "mean time to recover" and refers to the time needed to remove the cause for which the module had to be shutdown. If the module responds to the initiating event as expected, then shutdown heat removal capability should be provided for the period t_m. Therefore, t_m is the SHRS mission time if the reactor responds as expected. As it turned out, the SHRS failure probability depends significantly on this mission time.

The p-column in Table A4.1-1 represents the conditional probability of a specific initiating event occurrence given that an initiating event has occurred. In other words, it is the relative frequency of occurrence of a specific initiating event (relative to the frequency of occurrence of initiating events of all types). The p-column has been deleted.

The failure and repair rates of Table A4.2-14 are used as a part of the input to the FRANCALC 1 Code. The code estimates system unavailability using the FRANTIC-II code component unavailability models and the GE-developed PROBCALC Code models for evaluating system unavailability. A typical usage of these rates is the estimation of unavailability of a continuously monitored component which may be expressed as

$$\text{Unavailability} = \text{failure rate/repair rate}$$

Similar to FRANTIC II, input to the FRANCALC I code also includes description of the test frequency, test duration, efficiency of the testing and test-induced failures.

As stated in Section A.4.2-8, the SHRS failure probability was based on the assumption that pony-motor driven operation of the IHTS pump (block 137) is required to shutdown heat removal via BOP and ACS. The mission time of the IHTS pump is the time t_m defined earlier in this response.

AI-11 Comment

One basic assumption of the model in Figure A3.2-1 is that the intermediate states, e.g., accident types and core damage categories, are defined such that the subsequent analysis for a given intermediate state is independent of how such a state is arrived. For example, the plant may reach the core damage category 6 due to different initiating events. How can you be sure these initiating events have the same effect on the top events in the containment event tree for C6? Some initiating events tend to affect the top events in the event trees, e.g., seismic events, vessel rupture and station blackout.

Response

Some of the intermediate states of the model in Figure A3.2-1 are defined in terms of how these states were derived. For example, loss of flow at nominal coastdown rate with failure to scram is defined as of flow at nominal coastdown rate with failure to scram is defined as one of the accident types, some core damage categories are labeled with the letter "S" to indicate that SHRS should be considered unavailable in the event trees to be analyzed next, some release categories are defined as accompanied by early energetic release. In principle, therefore, there is no restriction on the model which prevents the definition of intermediate state from being expressed in terms of how these states were arrived.

As to the question of assuring adequate considerations of dependence on the initiating events and other system interactions, we have applied two procedures in the course of our work:

1. What may be called a "scanning procedure" whereby one develops an initial set of intermediate states and event tree top events, then scans through the whole model for each identified initiating event and system response which is suspected to create dependencies. It was this process that forced us, for example,to develop a different type of system event tree for IE-6, the large earthquake.

2. What may be called a "splitting procedure," whereby an intermediate state or category is broken into subcategories which reflect past history which would affect the consequences. Some discussion of this subcategorization is contained in Section A4.3.

It should be realized in the outset that the development of intermediate states, which adequately capture all parameters important for an accurate risk, is an iterative process with substantial input from the proper experts. The PRISM design offers a significant simplification of this process in that active systems are involved only in the areas of initiating events and system event sequences.

AI-12 Comment

It is not clear why some sequences that should have zero probability actually have non-zero probability, e.g., sequences with failure of RPS and PCS and successful pump trip in Figure A4.2-1.

Response

The probability value of 1.0 shown on some of the branches of the event trees is a rounded-off number which the ETA code produced on the plots. The actual inputs to the code were $1-\epsilon$ where ϵ is a very small probability number representing, for the example cited, the conditional probability of pump trip and which results in non-zero sequence probability as is shown on the right-hand column of each tree. This shortcoming of the ETA code will be avoided in future amendments of the PRA.

II. RVACS

AII-1 Comment

Why is failure of the RVAC system less probable for larger earthquakes? In general, why does the RVACS conditional failure probability (Table A4.2-15) depend on the initiating event when it is expected to be a diverse system?

Response

Table A4.2-15 has been revised to use more realistic mechanical failure probabilities consistent with the latest structural analysis conducted for PRISM. The revised table is attached. The last column in the table shows the conditional probability of RVACS failure given failure to remove heat via IHTS or BOP and given the initiating event. As seen from the table, the probability of RVACS failure increases significantly with the earthquake ground acceleration. Notice that a failure probability of 1.0 is used for RVACS if the earthquake causes these seismic isolators to fail.
Dependence of RVACS failure probability on the initiating event is due to two factors:

1) The revision time of RVACS operation. This depends on how long the module has to be shutdown to remove the cause which initiated the shutdown.

2) The initiating event may degrade RVACS capability. This is the case for other vessel failure (IE7, IE9) and RVACS blockage (IE21).

TABLE A4.2-15

CONDITIONAL FAILURE PROBABILITY OF SHRS

	Initiating Event (IE)	Via IHTS & BOP Given IE	Via RVACS Given Failure Via IHTS,BOP and IE	SHRS Failures
1.	Reactivity Insertion 0.07 to 0.18$	2.0E-3	6. E-10	1.2 E-12
2	Reactivity Insertion 0.18 to 0.36$	2.0E-3	6. E-10	1.2 E-12
3.	Reactivity Insertion $\geq$.36$	1.5E-2	4.4 E-9	6.6 E-11
4.	Earthquake 0.3 to 0.375g	6.0E-3	1.2 E-10	7.2 E-13
5.	Earthquake 0.375 to 0.825g	1.0E-1	4.4 E-7	4.4 E-8
6.	Earthquake >0.825g	1.0E+0	4.4 E-5*	4.4 E-5*
7.	Vessel Fracture	1.0E+0	4.4 E-7	4.4 E-7
8.	Local Core Coolant Blockage	1.4E-2	4.4 E-9	6.2 E-11
9.	Reactor Vessel Leak	1.4E-2	4.4 E-7	6.2 E-9
10.	Loss of One Primary Pump	2.0E-3	6.0 E-1	1.2 E-12
11.	Loss of Substantial Prim Flow	1.9E-5	1.6 E-11	3.0 E-16
12.	Loss of Oper Pwr Heat Removal	2.8E-4	8.6 E-11	2.4 E-14
13.	Loss of S/D Heat Removal via BOP	7.9E-5	4.8 E-11	3.8 E-15
14.	Loss of S/D Heat Rem via IHTS	1.0E+0	6.0 E-10	6.0 E-10
15.	IHTS Pump Failure	2.0E-3	6.0 E-10	1.2 E-12
16.	Station Blackout	2.9E-3	1.2 E-9	3.5 E-12
17.	NaH_2O Reaction IHX Failure	1.0E+0	4.4 E-9	4.4 E-9
18.	Spurious Scram and Transient Inadequately Handled by PC	2.0E-3	6.0 E-10	1.2 E-12
19.	Normal Shutdown	2.0E-3	6.0 E-10	1.2 E-12
20.	Forced Shutdown	7.9E-4	2.4 E-10	2.0 E-13
21.	RVACS Blockage	2.8E-4	1.0 E+0	2.8 E-4

* These valves apply when the seismic isolators function successfully. In case of isolator failure, these values should be replaced by 1.0.

AII-2 <u>Comment</u>

During a tornado strike and sudden depressurization of the RVACS inlet duct, couldn't the collector cylinder insulation be pulled off and potentially block the inlet duct? How will the insulation be held on for the module lifetime? Can the insulation burn (RVACS fire has been ruled out of Table A4.2-2)?

<u>Response</u>

The collector cylinder insulation is installed between metal cylinders, Figure 5.7-5, and is designed to safety related requirements. Specific insulation characteristics will be determined as the design developes. The RVACS fire parameter is measured by the PCS as discussed in Section 9.7.1.2. Table A4.2-2 is updated by Amendment 8 issued November 20, 1987.

AII-3 <u>Comment</u>

It is stated that RVACS would be needed less than one time per module life time (60 years). Total loss of offsite power occurs at a frequency of about one every ten years of LWRs. With no safety related systems (other than RVACS) available to remove decay heat during a loss of offsite power, isn't once in 60 years optimistic?

<u>Response</u>

The shutdown heat removal system reliability analysis shows that the usage frequency of the RVACS is less than once during the life of the plant. In this analysis the characteristics of the PRISM power plant with three separate power blocks were highlighted in comparison to a large monolithic plant. In a loss of offsite power event two power blocks are shut down and the third block kept running at low load to take care of outside requirements such as operating the normal decay heat removal systems for the two shutdown power blocks. The design of the outside electrical power system is such that this interconnection is a normal situation. If this power supply is also lost for some reason, there is the backup ACS which also operates entirely by natural circulation, i.e., natural circulation in the intermediate sodium loop and natural convection of air at the steam generator surface after manual opening of the air flow louvers. Thus, this system also requires no power, onsite or offsite, to perform its function of cooling the plant. The need for RVACS arises when the sodium in the IHTS has been lost either by a major pipe break or a steam generator rupture disc failure.
The Auxiliary Cooling System (ACS) is used to supplement RVACS during a loss of offsite power. The ACS limits the reactor core outlet temperature to about 925°F maximum and reduces the reactor cool down time to hot standby temperature from 80 days (with RVACS alone) to about four days. IHTS sodium flow during this cool down is provided by either the IHTS pump pony motor (which receives power from the standby gas turbine) or by natural circulation. Although the ACS is not a safety related system, it is used to supplement RVACS during abnormal conditions such as loss of plant electrical power to reduce the severity of the plant transient conditions and increase plant

availability by avoiding the long plant cool down which would occur with RVACS alone.

AII-4a **Comment**

Please list all possible failure modes of the RVACS.

Response

All failure modes for the RVACS are grouped into, a) loss of primary sodium natural circulation, b) loss of RVACS air natural circulation and c) loss of heat transfer between the primary sodium and RVACS air. The primary sodium circulates up through the core to the hot pool above the core, a) into the annulus between the vessel wall and liner cylinder and b) into the inlet of the two IHX's, down to the cold pool around and below the core, up through the fixed shields around the core, into the inlet plenum for the four EM pumps and through the eight pump discharge lines to the inlet plenum for the core as shown in Figure 5.7-2. The RVACS air circulates through the four cold inlet ducts down to the cold plenum above the silo, down the outer annulus between the cold collector cylinder and silo wall, around the lower edge of the collector cylinder, up the inner annulus between the containment and the hot collector cylinder to the hot plenum above the cold plenum and up the four hot outlet ducts as shown in Figure 5.7-2 and 5.7-3. The core heat transfers from the primary sodium to the reactor vessel, the argon filled annulus between the reactor and containment vessels, the containment vessel to the RVACS discharge air as shown in Figure 5.7-5. The RVACS failure modes are all those that allow the core to heat beyond design limits by impeding or blocking the circulation of the sodium or air or the heat transfer from the sodium to the air. The fault tree for the RVACS failure is shown in the figure that follows.

AII-4b Comment

Will construction of the RVACS collector cylinder be covered by ASME code? Has failure/degradation of the collector cylinder been investigated? Is it possible to set up natural circulation loops within the RVAC system which could be ultimately worse than simple blockage?

Response

The construction of this safety grade, seismic category I component will be in accordance with Section III of the ASME code.

With regard to the second part of the question, extensive efforts to determine failure/degradation mechanisms of the collector cylinder have not been performed. However, it should be noted that the structure is not loaded significantly. It mainly has to support its own weight. Also, with the horizontal seismic isolation system in PRISM, any seismic loads are significantly reduced. Thus, the potential for structural failure of the collector cylinder is considered very low. The effects of various postulated failure modes, e.g., flow blockages at various locations, structural failure of the collector cylinder etc. have not been performed either.

At this time we cannot address the third part of the question. However, many types of failures/blockages can be accommodated because the cold air tends to find its way to replace hot air which tend to rise by natural circulation. For instance, blockage of all air inlets may be acceptable because cold air down flow and hot air upflow zones may be established around the circumference of the containment vessel. Similarly, failure of the collector cylinder at the top such that it drops to the bottom and rest of the reactor silo floor would result in similar air flow paths. These postulated failures have not been investigated yet, but 3-D computer codes to do such analyses have been identified. There are no specific plans to perform such analyses and/or any supporting experiments at this time, but work in this area will be required in the future.

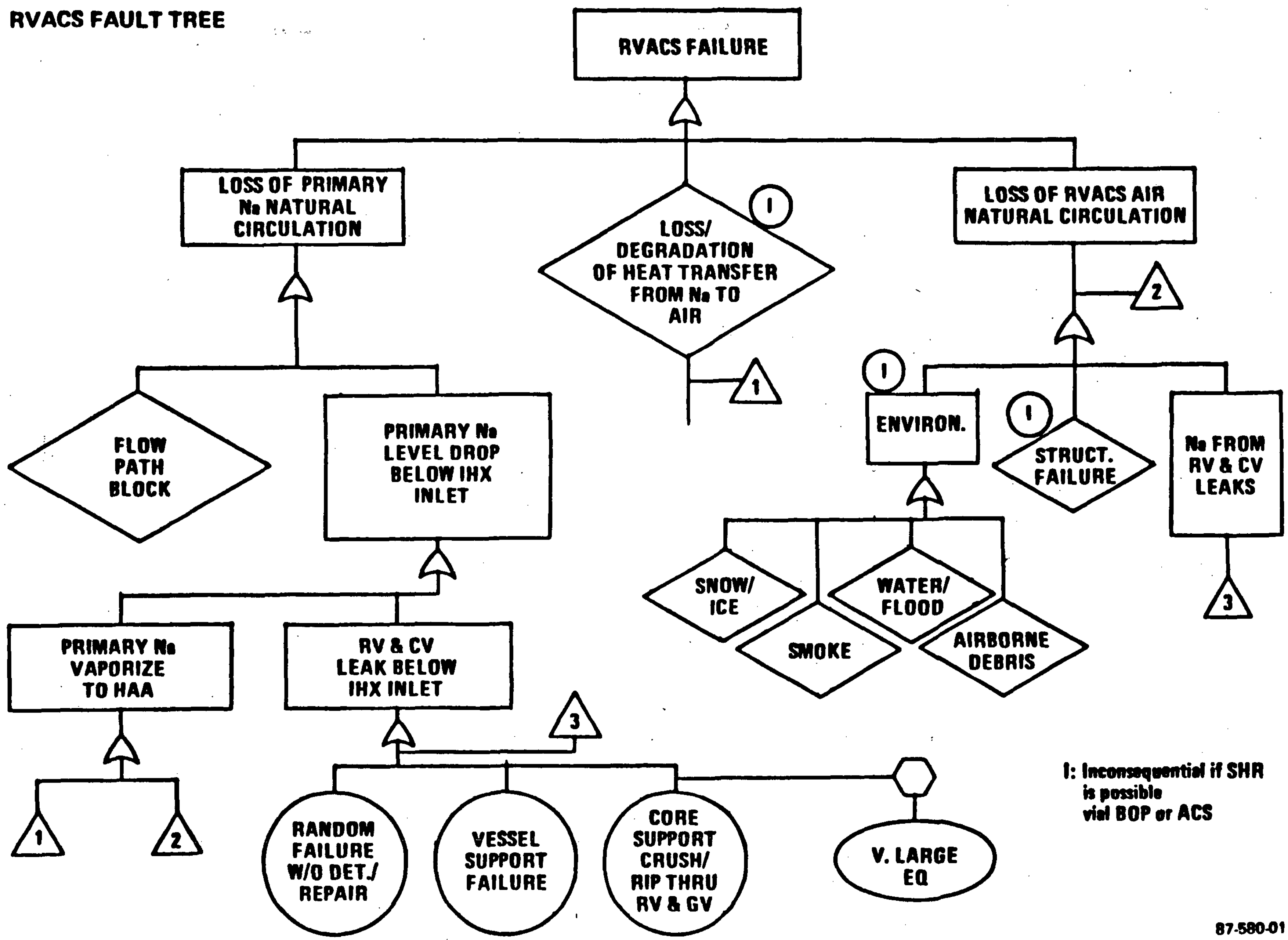
AII-4c <u>Comment</u>

Should be RVACS become blocked and temperatures inside both the inlet and outlet duct become equal, can natural convection be reinstated after blockage removal? If so, in what time period?

<u>Response</u>

This particular case has not been analyzed but it is believed that natural convection of air will restart after blockage removal. The reason for this is that the cold and hot air annuli are separated by the insulated collector cylinder. This will continue to maintain a significant temperature difference between the two regions. The hot air will rise because of buoyancy and escape and be replaced by colder air. This process is anticipated to take place within minutes, but completely normal operation of the system to be established is expected to take hours because the collector cylinder and concrete silo wall will be heated up and these structures, particularly the concrete, has a large time constant.

RVACS FAULT TREE



AII-5a Comment

Why is IE21 considered an initiating event? Does normal power operation require RVACS?

Response

IE-21 is defined as RVACS blockage. Such a blockage presents system degradation which should be removed. If the blockage is benign with no buildup, the operator may postpone removing the blockage until the next scheduled outage. If the blockage is sudden, say following an extreme incredible sandstorm, the plant may be brought to forced shutdown to renew the shutdown heat removal capability.

In regards to the second question, normal power operation does not require RVACS, but the RVACS, as a totally passive system, is always in operation.

AII-5b Comment

It is stated (A4-1) "By definition, all initiating events require the module to shutdown." Wouldn't shutdown, given INITIATING EVENT 21 (RVACS Blockage) put the module in a less safe condition?

Response

The PRISM reactor has three diverse paths for shutdown heat removal; via BOP, via IHTS, and RVACS. Naturally, sutting down the reactor to restore RVACS in case of blockage is done while the other two heat removal paths are available.

Please refer to response to Comment AII-5a for further discussion of the reason and timing of reactor shutdown given IE-21.

III. VESSEL

AIII-1a Comment

In analyzing initiating event 7, vessel failure, is the probability of reactor vessel failure taken as 10^{-7} or 10^{-13}? Is the probability of containment vessel failure considered? What value? What is the resulting contribution to the probability of prompt fatalities?

Response

The probability value used for reactor vessel fracture defined by initiating event 7 is 10^{-13}/yr. As stated under the definition of initiating event 7 in the PSID, this probability has been assigned by judgment based on qualitative fracture mechanics considerations and comparison of PRISM vessel to other nuclear vessels for which probability estimates of 10^{-11}/yr were evaluated.

The probability of the containment vessel is not included in the above estimate of reactor vessel probability. However, the probability of containment vessel failure has been considered in the RVACS reliability assessment following an accident. Preliminary stress analysis indicates that the impact of reactor vessel drop on the containment vessel will result in minor stress (5400 psi). The probability of containment vessel failure under such an impact has been estimated as 4.4×10^{-7} given the reactor vessel fracture of initiating event 7.

The contribution of initiating event 7 to prompt fatalities is estimated as $\sim 10^{-15}$ per module year in the one mile radius for which acute fatalities have been estimated using the MACCS code. This contribution amounts to less than .002% of the total acute fatality risk estimated.

AIII-1b <u>Comment</u>

The PRISM reactor vessel design appears to have substantially lower stresses than typical LMFBR vessels? Please explain why it is two orders of magnitude lower than the failure rate of E-11/yr.

<u>Response</u>

Structural performance of the PRISM reactor vessel has been evaluated for thermal and mechanical loads during normal operation, thermal transients, OBE and SSE seismic loads and the extended operation under RVACS cooling with loss of other heat sinks. Results of the evaluation have shown substantial margins against fatigue and creep failure, thanks to the low primary stresses and operating temperatures. For example, the ASME Code allows more the 10^6 cycles for the levels A&B stresses calculated for PRISM reactor vessel, against less than 2000 expected cycles in the duty cycle.

The above structural margin, the ductile vessel material which ensures leak before break if a leak ever occurs, the leak detection system reliability, and the incredibly large critical leak size required for vessel rupture all add up to ensure confidence of a highly reliable vessel. Moreover, PRISM offers two unique characteristics which have been credited with the two orders of magnitude reduction in failure probability indicated in the comment.

1) Factory manufacturing and pre-operational testing. This allows for better environment for welding, heat treatment, inspection, and QA than offered if these processes were done on-site.

2) Lack of vessel penetrations and stress raiser such as nozzles and pipe connections. As a pool reactor the PRISM vessel has no loop piping. Moreover, instruments, cover gas and sodium purification system lines, etc. are all connected through the vessel cover.

AIII-2 Comment

Ref. p A4-16. It states that "gross structural failure of the vessel will occur and core meltdown is not unlikely" for seismic >.825g. The probability of failure given this event however, is 1.3 E-3. Isn't there a contradiction?

Response

The statement "gross structural failure of the vessel will occur and core meltdown is not unlikely" has been made in reference to the case if seismic isolation of the reactor vessel fails. That is, given that seismic isolation fails, then the indicated consequences follow. On the other hand, the probability of 1.3E-3 refers to the probability of seismic isolation failure given the large earthquake.

IV. SEISMIC

AIV-1 Comment

The information/documentation provided on the seismic analysis is very limited. Please provide: a) The range of seismic events analyzed and why more ranges were not specifically considered. Please explain the earthquake range covered by IE-4. b) The seismic events that were considered in evaluating Table A4.2-9? c) The report on the assessment of component failure under SSE loading that was used in evaluating Equation 4.2-18. d) The analysis in assessing the probability of gross vessel failure due to a seismic event.

Response

a) Seismic events considered for the PRISM PRA cover the spectrum of ground peak accelerations from 0 to 1.2g. The GESSAR-site seismic hazard curve (frequency per year of exceeding a given acceleration) has been used in accordance with the scope defined in Section A1.1, and as shown in Table A3.2-1. As described in Section A4.1.2, the above spectrum of ground acceleration has been divided into five ranges which lead to different plant responses. These are

0 - .15g (OBE):	In this range the PCS continues plant operation.
.15g - .3 (SSE):	In this range the PCS shuts down the reactor in an orderly manner using automatic or manual fast power runback.
.3g - .45g:	In this range the RPS trips the reactor if any of its setpoints is reached despite the expectation that the plant systems will not suffer any significant structural damage.
.45 - .82g:	In this range some damage to seismic category I and non-safety grade equipment may occur, but the seismic isolation system is expected to prevent any structural damage to isolated equipment and structures. The RPS will trip the reactor and RVACS will remove the decay heat.
>.82g:	This range is at the edge of the isolation system capability based on the horizontal gap between the isolated and non-isolated structures and damage to isolated equipment and structures can occur if the seismic gap closes. For an earthquake above 1g some damage may occur from vertical motion.

An earthquake in the first range (0 - .15g) is not considered an initiating event since it does not lead to reactor shutdown. An earthquake in the range (.15g - .3g) is considered a contributor to the group of forced shutdown events, initiating event 20. The remaining three ranges have been defined as initiating events 4, 5 and 6 in section A4.1. Section A4.2 provides an assessment of the failure probabilities of plant systems given any of the initiating events.

b) The median seismic capacity and standard deviation of Table A4.2-9 were used to estimate the fragility of the primary pump coastdown system given earthquakes of .3g, .6g and 1.2 g. The results are shown in the fault trees in Figures A4.2-36 through A4.2-38. Table A4.2-9 has been amended to contain this information and is attached for reference.

c) Equation 4.2-18 was based on the generic equipment fragilities reported in UCRL-92798 (James E. Wells, "Reliability Data Required for A Seismic Risk Assessment," LLNL, Dec. 1985). A GE-computer Code used this data to estimate failure probability of RPS components at 0.3g. Relays showed the highest failure probability given a SSE. The estimate obtained was 1.11×10^{-3}. Using this failure probability, and the conditions that failure of 3 relays or more out of 4 constitutes a circuit failure, leads to the following probability of relay subsystem failure:

$$P = \left(\frac{4}{3}\right) (1.11 \times 10^{-3})^3$$

$$= 5.47 \times 10^{-9}$$

Equation 4.2-18 approximates the above values as

$$P \sim 6 \times 10^{-9}$$

It should be noted that Equation 4.2-18 is outdated. New reliability models and estimates which are consistent with current RPS design have been developed. The new assessment is included in Amendment 8.

d) The analysis of the seismic vessel failure is included in Amendment 8 issued November 20, 1987.

TABLE A4.2-9

DATA USED FOR PUMP COASTDOWN RELIABILITY EVALUATION

	Equipment Fragility		Failure Probability (per demand)			
				Seismic Initiator		
Component	Median Capacity	Standard Deviation	Nonseismic Initiators	.3g	.6g	1.2g
EM Pump	8.9 g	0.65	2.2×10^{-7}	2.2×10^{-7}	2.2×10^{-7}	2.2×10^{-7}
Synchronous Machine	2.1 g	0.65	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}
Circuit Breaker	4.1 g	0.65	5×10^{-8}	5×10^{-8}	5×10^{-8}	5×10^{-8}
Regulator	2.72g	0.65	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}
Housing (Structure)	1.1 g	0.45	~0	~0	1.55×10^{-3}	6.6×10^{-1}

AIV-2a Comment

Seismic isolation is dependent on horizontal soil movement. Please provide information (graph) on horizontal movement verses g which is needed to determine adequate seismic isolation separation.

Response

Seismic isolation is achieved by introducing flexible (seismic bearings) between the isolated structure and the ground. This reduces the horizontal fundamental frequency to a range of 0.5 to 1.0 Hz which is well below the range of harmful frequencies for most earthquakes. The added flexibility, required to reduce accelerations and forces results in relative displacements between the isolated structure and the ground. The magnitude of these displacements depends on several factors including the fundamental frequency of the bearings, the damping of the bearings and the ground motion characteristics.

In the PRISM design the isolators have a horizontal frequency of 0.75 Hz and a damping factor of 10%. The maximum horizontal displacement computed based on the 0.3 g safe shutdown earthquake (SSE) and a time history enveloping the NRC RG 1.60 spectrum is 7.5 inches. This includes a combination of the two horizontal directions and torsional effects. A horizontal separation of 20 inches is provided between the isolated and fixed reactor facility structures to provide an adequate margin for consideration of larger earthquakes. For larger earthquakes, the displacements can be increased linearly as a function of increased accelerations. This approach assumes that the RG 1.60 spectra are valid for all earthquakes which is overly conservative particularly for very large (greater than 0.6g) earthquakes. For these cases a site specific design earthquake will have to be developed; however, it is expected that the relative displacement for a 0.6g earthquakes will be less than 15 inches.

AIV-2b Comment

In addition, as we go higher in earthquake force (g), what will be the first major component failure that would ultimately lead to a core damage state?

Response

The first component failure that would produce core damage is the failure of the seismic isolation bearings. The expected capacity of the current bearing design is about 1g SSE peak ground acceleration as opposed to the PRISM design basis of 0.3g SSE peak ground acceleration.

Seismic loads could cause core damage through:

1) direct structural damage due to inter-assembly impact under lateral seismic excitations,

2) failure to scram because of damage to the control drive/latch system,

3) unacceptable reactivity insertions from seismic core compaction under horizontal excitations or core/control rod separation from vertical excitations.

Analyses indicate the design basis seismic loads transmitted to the reactor module and internal components to be insufficient to initiate any of these failure modes. This is a consequence of the filtering action of the seismic isolators which are designed to produce a lateral system frequency of 0.75 Hz and which preclude any significant spectral amplifications of the horizontal seismic excitations at frequencies greater than 1.5 Hz. In comparison, the minimum reactor component frequency is 3.6 Hz. Thus, the first component to fail which would lead to core damage is the seismic isolators.

V. SHRS

AV-1 Comment

Regarding the event tree in Figure A4.2-6, given the successful seismic isolation, failure of RSS to insert rods, pump trip, and failure to coast down, doesn't sodium boiling and a potential power excursion result? What significance does decay heat removal have under these conditions?

Response

As delineated in Figure A4.2-6, the indicated accident sequence leads to accident type "G3" with SHRS available or "G3S" if SHRS is lost. According to Table A4.2-1, accident type "G3" is a combined severe unprotected loss of flow and transient overpower accident which is either: 1) a flow coastdown at nominal rate accompanied by reactivity insertion equivalent to withdrawal of all the control rods, or 2) a flow coastdown at a rate faster than nominal accompanied by a reactivity insertion less severe than the withdrawal of all control rods. The core response event tree for accident type G3 is shown in Figure A4.3-11. The event tree shows that shutdown before significant damage is likely, thanks to the PRISM inherent characteristics. Figure A4.3-11 shows that accident type G3 leads to an energetic excursion, such as described in the comment and which damages the reactor vessel, in only 5.5% of the cases (sequence class C_6 in Figure A4.3-11). Consequently, the SHRS availability to remove residual and decay heat is important for continued radioactivity retention after neutronic shutdown which may occur with the core virtually intact or partially changed.

AV-2 Comment

Is IE14 conditional on a scram? Is IE15 an event that causes IE14?

Response

The initiating event IE-14 is a loss of shutdown heat removal via IHTS. As stated in Section A4.1, the dominant failure mode for the event (which includes leaks from the pump housing and seals) is an IHTS leak. As an initiating event, the leak is assumed to occur during full power operation. Therefore, it is not conditional on scram.

On the other hand initiating event IE15 is defined in Section A4-1 as failure of the IHTS pump. The dominant failure mode which was assumed for IE15 in the PRA was failure of the main pump motor which would not fail the pony motor function in the shutdown heat removal mode.

AV-3 Comment

What kind of SHRS restoration during the grace period following an earthquake >.825g can be accomplished such that it would prevent fuel failure?

Response

The kind of SHRS restoration following an earthquake >.825g will naturally depend on the failure mode. The PRISM SHRS has three diverse paths for heat removal; via BOP, via IHTS, and RVACS. Assuming that all three paths have failed, then a reasonable strategy for recovery would be to repair the heat removal path that can be recovered in the shortest possible time. If that path is RVACS, then the most likely failure modes would be blockages of the air passages or inlet gates by structural debris, or degradation of the heat removal capability of heat transfer surfaces in case of a sodium fire. Clearing the air passages and cleaning the heat transfer surfaces from sodium aerosol will be the kind of restoration used in this case.

AV-4 Comment

In the sequences involving loss of shutdown heat removal (R6 and R8 Release Category Events), a slow process of sodium heatup and boiling occurs. When the core is predicted to uncover, core melting occurs. The analysis apparently assumes that at this point the core falls into the lower plenum and guard vessel failure ensues. The fission products are assumed to remain with the sodium from where they are removed as boiling and burning of the sodium proceeds. It appears conceivable, however, that if melting occurs gradually as the sodium boils off then the fission products will be released to the gas in the upper plenum rather than to the primary liquid sodium. These fission products would then be available to the HAA considerably earlier in the accident sequence. What are the consequences of this sodium "bypass" route for fission products and the potentially earlier release of fission products to the environment?

Response

In the sequences involving loss of shutdown heat removal (R6 and R8) the impact of using the assumption that fission products produced during core melt are released to the sodium vapor/cover gas phase rather than to the liquid sodium is primarily on the release of volatiles and especially Iodine. Neglecting condensation of NaI on surfaces in the Head Access Area, the release of Iodine from the vapor phase would be analogous to released noble gases. As can be seen in Figure A4.4-16 for category R6A there is only a minor enhancement in the release timing of halogens (curve D) by using that for noble gases (curve A). Thus the difference in the effect on calculated prompt fatalities should be negligible. As can be seen in Tables A4.5-1 and A4.5-2 there are essentially no prompt fatalities for these releases. This is true despite the fact that the original assumption of release of Iodine in direct proportion to sodium boiled-off is conservative. It is well-known that Iodine is actually released late in the boil-off process.

VI. RPS

AVI-1 Comment

Figure A4.2-27 - Does failure of 2 RPS channels and one scram breaker lead to failure to RPS?

Response

The PRISM RPS has four identical divisions (from sensor through scram breakers) with a fault tolerant reconfigurations capability. This makes the RPS able to identify and isolate a faulty component of any division, allow the component to be out for repair or service, while continuing to use the healthy components of the division. This reconfiguration capability has been modeled in the RPS reliability model by a system which is redundant at the component level. Therefore, any of the following failures formed a necessary condition for RPS failure as modeled in the PRA: (1) three out of four sensors, (2) three out of four logic trains, (3) three out of four breakers. Consistent with the above conditions, Figure A4.2-27 does not imply that failure of 2 RPS channels and one scram breaker lead to failure of the RPS.

It should be noted that the RPS design has changed significantly since Appendix A was issued. The current design utilizes a different set of sensors, 12 scram breakers for the control rod latch mechanisms (2 breakers per rod), and 12 scram breakers for the control rod drive-in mode. Moreover, operation of the system no longer allows two divisions to be out while the reactor is at power. The reliability model of the new RPS design is included in Amendment 8 issued November 20, 1987.

AIV-2 Comment

Does equation 4.2-13 imply that three RPS channels are under maintenance simultaneously? Is there a tech. spec. that limits the number of channels under repair? Do you consider failure on demand of the components in RPS? Figure A4.2-27 shows that event 1 leads to failure of RPS. Why are some probabilities in Table A4.2-5 lower than the probability of event 1? Why don't you consider common mode failure of the scram breakers?

Response

In accordance with the RPS characteristic number as stated in p. A4-30, the RPS tech spec was assumed to allow operation while two channels were out for maintenance or repair. Equation 4.2-13 models the situation where two channels are both out for maintenance or repair, during which time a third channel fails.

Current RPS design does not allow operation with two channels out at the same time. The new RPS design will trip the reactor if a channel fails when another one is out for maintenance or repair.

Only time-dependent failure of electronic components of the RPS has been used in this PRA. Constant (on demand) unavailability has not been considered.

As correctly noted in the comment, Event I is a failure cause for the RPS. Table A4.2-5, however, presents the conditional probability of failure of both the RPS and PCS. As stated in Section A4.2.2-4, the results presented in the above table do not take credit for manually-initiated scram.

The question of common mode failure for the scram breakers and the RPS components is addressed in Amendment 8 issued November 20, 1987.

AVI-3 Comment

Page A4-42 - Please explain the expected responses of the operating power heat removal system to different initiating events. What does the operator need to do?

Response

The normal expected system response to three initiating events is described below. These events are 1) Reactor Protection System(RPS) scram resulting from a high reactor module flux, 2) RPS scram

initiated by low primary sodium flowrate, and 3) Reactor module fast runback initiated by a sodium-water reaction.
1) RPS scram on a high reactor module flux:

High flux level in a reactor module (currently 112% of rated) will initiate RPS scram in the affected module. The RPS will also trip the affected modules' primary sodium pumps, initiating primary sodium flow coastdown. These RPS operations are automatic. Concurrently, the Plant Control System(PCS) will determine that continued operation in the turbine-leading-reactor control mode is not feasible due to the rapid loss of one of the three reactor modules in the affected power block. The PCS will switch to a reactor modules-leading-turbine control mode in the affected power block. The PCS will trip the sodium pump in the affected intermediate heat transport system within the first second following scram. During the power reduction, the PCS will also automatically trip the recirculation pump in the affected steam generator recirculation loop. With the reduction of steaming rate in the affected steam generator, its feedwater control valve reduces feedwater flow to maintain a constant steam drum level. Control is eventually automatically transferred to the smaller startup feedwater control valve. Since the turbine in the affected power block is now following the rapidly changing reactor heat generation, its control valve position is controlled to maintain a constant steam header pressure. The electrical power generated in the affected power block will reduce from rated to approximately two-thirds of rated.

For the PRISM saturated steam cycle, the steam temperatures are only a function of pressure. Since the steam generating systems are at essentially the same pressure, their steam temperatures are also nearly identical. Therefore the tripped modules' steam generating system is not required to be isolated from the main steam header following scram. These PCS control actions are automatic, and no operator action is required in the affected power blocks' transition from full power to two-thirds of rated power following a reactor module trip from high flux.

2) RPS scram on low primary sodium flowrate:

Low primary sodium flowrate in a sodium module will initiate RPS scram on a calculated module power to flow ratio of 1.4. Measurements of primary sodium pump outlet pressure and reactor flux are used to calculate this ratio. Once RPS scram is initiated, the sequence of RPS and PCS actions is the same as for the first case, RPS scram on high reactor module flux. Again, for this case all RPS and PCS actions are automatic and no operator action is required in the affected power blocks' transition from full power to two-thirds of rated power.

3) Reactor module fast runback initiated by a sodium-water reaction:

Response to a sodium-water reaction could vary from an operator initiated slow reduction of affected module power (over a period of hours) for a very small leak, to a rupture disk failure with subsequent intermediate heat transport system sodium drain and automatic PCS directed isolation and blowdown of the affected steam generation system for a large sodium-water reaction. Actions for a limiting design basis sodium-water reaction will be described. The event will consist in an instantaneous rupture of at least one steam generator tube during full power operation.

The event will result in failure of the rupture disks which will initiate PCS plant investment and control actions. The PCS will initiate automatic isolation and blowdown of the steam generation system, including trip of the associated recirculation pump, terminating the reaction. The intermediate loop sodium is also drained without any required operator actions. The intermediate heat transport system sodium pump is automatically tripped by the PCS to avoid possible mechanical damage. The PCS also initiates automatic fast power runback of the affected reactor module. The module will be taken subcritical in a little more than two minutes. The primary sodium flow is reduced proportionately, under PCS control. Should the primary cold leg sodium temperature reach 800 degrees F° before the module is subcritical, a RPS scram will take the module subcritical within seconds and initiate primary sodium flow coastdown. The power block control mode will also switch to reactor modules leading the turbine on the large heat generation to power generation unbalance.

The turbine admission valve position is controlled to maintain constant main steam header pressure during the event. Power is reduced from rated to two-thirds of rated, as in the previous events. Again, no short term operator actions are required, although manual intervention in the sequence of PCS actions by the power block operator is permitted.

AVI-4 <u>Comment</u>

Explain why the conditional failure probability of the PCS and RPS is so low in Table A4.2-5 (IE-6) given the catastrophic INITIATING EVENTS 6 and 7.

<u>Response</u>

The conditional probability of PCS and RPS failure given IE-6 has been assigned by applying a subjective degradation factor to the probability given an SSE. The latter probability of failure has been estimated using seismic fragility data for generic components. As indicated in the response to Comment VIa-6, later investigation indicated that the subjectively assigned degradation factor may have been too optimistic for IE-6.

As for IE-7, the conditional failure probability was obtained by applying a subjective degradation factor to the failure probability given a large reactivity ramp (IE-3). This degradation factor reflects the concern that an incredible event such as the vessel fracture of IE-7 will almost certainly reflect a collapse of nominal administrative controls and technical competence in the design, manufacturing, construction, QA, ISI or operation of the plant. We believe that the degradation factor used has been conservative.

As noted in many other comments, the RPS and PCS designs have changed after Appendix A of the PSID was issued. Revised failure probability estimates reflecting these changes will be included in Amendment 8 issued November 20, 1987.

AVI-5 Comment

Ref. p A4-52. How is IE 21 picked up by the RPS or PCS?

Response

Initiating Event 21, RVACS blockage, is picked up by RPS mass air flow (Class 1E) sensors in the RVACS exhaust. This data is automatically fed to the PCS. (See the response to Comment 5.19).

AVI-6 Comment

Explain why the failure probability of the "nominal inherent reactivity feedback system" is assigned a value of 10**-6 under the condition that the "signal to for shutdown fails" to occur. Why is this different from the failure probability value of 0.1 assigned to the same failure mechanism under the condition that "enough control rods inserted by RSS" has failed?

Response

Section A4.2.6 of the PSID, titled, "Inherent Reactivity Feedback System Reliability," provides the definition of the inherent feedback system and explains the basis for the probability of its failure for various initiating events and reactor shutdown system (RSS) responses. In accordance with this section, "the control rods must be able to move in their guide tubes and the fuel subassemblies must be able to move against the core restraint system," for the core to provide negative reactivity feedback when the primary sodium heats up. As stated in Section A4.2.6, fault tree analysis of the RSS shows that stuck control rods account for 10% of the RSS failure probability. It has been assumed in Section A4.2.6 that the inability of the control rods to move constitutes failure of the inherent shutdown system. Therefore, a value of 0.1 was assigned for the conditional probability of the inherent feedback failure given failure of the RSS. On the other hand, a probability value of 10^{-6} has been assigned by judgment to reflect the incredibility of the inherent feedback failure for other situations which do not include seismic events or reactor vessel failure. Section A4.2.6 assigns higher failure probability values for these latter events.

VII. EM PUMPS

AVII-1 Comment

What is the difference between IE11 and IE16? How does an EM pump in one power block receive power from another power block? Is it automatic? Do you take any credit for the standby generators?

Response

Initiating events IE-11 and IE-16 as defined in Section A4.1 were not mutually exclusive. Subsequently, IE-11 has been redefined as "loss of electric power to two primary pumps simultaneously," i.e., IE-11 no longer includes the loss of off-site and on-site power. The definition of IE-16 has been retained as "station blackout: which is defined as "loss of all off-site and on-site power sources capable of running the BOP, IHTS and primary pumps." The new definition of IE-11, its frequency of occurrence, mean time to recover, and analysis is contained in Amendment 8 issued November 20, 1987.

The main single line electrical diagram (Figure 8.3-1) shows the connections of the 3 power blocks to each of the two buses for the common station service system. EM pump power can be supplied automatically from the buses as shown on Figure 8.3-5. The synchronous machine trip to pump coastdown (failsafe) if the pump power supply is interrupted. No credit is taken for the standby generators.

AVII-2 Comment

Can the synchronous machines be taken out for maintenance? Can they be shut down if they fail, catch fire, etc?

Response

The synchronous machines are located in separate divisional vaults and can be controlled individually. Taking a machine out of service will stop the associated EM pump. Maintenance is normally scheduled when the reactor is shut down. Reactor operation on three pumps is not a normal mode.

VIII. POWER SUPPLY

AVIII-1a Comment

More information is required on the electric power system that feeds the systems that are important to the PRA, e.g., RPS, PCS, control rod drive motors, EM pumps, intermediate coolant pumps, feedwater pumps, and condensate pumps.

Response

Chapter 8 contains the description of the electric power systems. Figure 8.3-1 is the overall plant single line diagram showing the offsite and onsite redundant power sources. Sheet 2 of the figure shows the 7.2 Kv power source for specific as well as typical pump motors. Figure 8.3-1, 2 and 3 show the redundant and uninterruptable AC and DC power for Class 1E equipment (i.e., RPS and control rods) and Figure 8.3-5 shows the power to the EM pumps. The building electrical power system, Figure 8.3-7, includes the redundant and uninterruptable non-class 1E power to the PCS. This figure is incorporated into Chapter 8.

AVIII-1b Comment

Why has loss of all DC not been considered an initiating event? Do we understand the design well enough to rule out this event?

Response

The loss of all DC is unlikely as there are a minimum of two independent, battery backed DC sources for each DC system. Each latch coil has two parallel circuits sourced from two different DC supplies. Each control rod drive-in motor has two different sources of DC voltage. Each of the four divisions of the RPS has its own, battery backed DC power, etc.

The loss of all AC power to a reactor module is the initiating event. The EM pumps will automatically go into a coastdown flow with energy supplied by the synchronous machines. The RPS will sense a decrease in flow and initiate a reactor trip sequence. It is also noted that the latch coil DC power supplies are purposely designed with a limited capacity such that the rods will drop automatically after a brief delay following an AC power outage (the delay is designed to prevent scram on AC power line transient events).

AVIII-2 Comment

The PSID states that (8.3-1) "in the event that secondary off-site power is lost, the generators will furnish power to common equipment loads essential to maintaining plant operation and preventing major equipment damage." What major equipment damage can be expected should the generators fail?

Response

No safety related equipment would be damaged if the auxiliary generators fail. However, the main turbine generators may be damaged. After a trip the turbine generator is shut down in a controlled mode to preclude shaft bowing, thermal stress etc. damage. Power is therefore required to be supplied to the main turbines auxiliaries such as lube oil pumps, turning gear motors, service motor pumps, and the scaling system.

IX. BOP

AIX-1 Comment

Most of the initiating events in light water reactors are in balance of plant. As LMR balance of plant is similar to LWR balance of plant, shouldn't LMR's experience almost as many initiating events per year? (This would be 5-10 events per year, where PRA total is more like one event per year.)

Response

The PRISM project position regarding BOP-initiated transients has been to minimize their frequency of occurrence for the sake of safety and plant availability. The BOP and Plant Control System (PCS) were to be designed to control as many such transients as possible to avoid the need to shutdown or to scram. This is reflected in the small number of transients used in the PRA. However, overriding considerations to design the NSSS so that it is as independent of the BOP as practicable has led the Project to define as part of the PCS duty cycle all BOP transients experienced in current LWR reactors. The PCS is to either control the plant parameters to avoid shutdown if possible, or bring the reactor to an orderly shutdown configuration using the PCS fast runback features. Consequently, a frequency of BOP transients of 5.5/year has been defined for the update PRA contained in Amendment 8 issued November 20, 1987.

AIX-2 Comment

A typical value for the frequency of SG tube rupture for an LWR is 3×10^{-6} per year. Why is the frequency for PRISM much lower?

Response

This comment seems to be referring to IE=17, very large Na-H_2O reaction. The event is defined as "a very large sodium water reaction in the steam generator." A frequency of 6×10^{-8}/year has been assigned to the event. Although the frequency of IE17 is much lower than the quoted LWR frequency of tube failure, we believe that the PRA assigned a conservative frequency for IE17. The following discussion provides the rationale for why IE17 should be expected to occur less frequently than actually assumed in the PRA.

Worldwide PWR steam generator tube failures for the years 1983 and 1984 have been analyzed in the Nuclear Engineering International issue of June 1986. The experience included 4 million tube years of operating PWRs SGs. Statistics of the failure causes indicate that failure of the PWR steam generator tubes has been dominated by:

1) Stress corrosion cracking by inter-granular attack on the secondary side. This is responsible for 46.3% of the failure.

2) Stress corrosion cracking in the primary side - This is responsible for 26.8% of the failures.

GE material experts believe that none of the above failure mechanisms will be significant for the ferritic steel (2-1/4Cr, 1Mo), the range of temperature, and the sodium environment considered for the PRISM steam generator. Moreover, the PRISM design concepts currently under investigation emphasize the requirement for tube-tube sheet weld inspection and heat treatment. Based on these factors, it is estimated that the PRISM steam generator will fail at a rate which is a factor of 10 to a factor of 16 lower than the rate experienced for LWR's. Subsequently, the PRISM project made it a requirement that no steam generator tube failure shall occur in a plant of 9 modules during its service life of 60 years.

The design basis sodium-water reaction considered for PRISM assumed one tube to have a sudden double ended rupture followed by two double ended tube ruptures one second apart. This kind of extreme mode of failure for the ductile steel considered for the PRISM steam generator will be expected to occur at a lower frequency than more common leaks. Therefore, for IE17, which is defined as a sodium-water reaction equal to or larger than the design basis sodium-water reaction, one would expect the frequency to be at least two orders of magnitude less than the quoted LWR frequency, i.e., IE17 frequency in the order of 10^{-8}/year.

AIX-3 Comment

For initiating event 13, loss of shutdown heat removal via BOP, is it implicitly assumed that there is a scram (this would contradict Figure A4.2-13)? If there is no scram, can BOP with one condensate train and one feedwater train function to remove "shutdown" heat?

Response

As defined in Section A4.2-1, loss of shutdown heat removal via BOP means loss of both of the two condensate pumps and the three feedwater trains. As an initiating event, the event is assumed to occur while the reactor is operating at full power, and hence is consistent with Figure A4.2-13. However, the same event presents also a loss of one of the paths for shutdown heat removal as discussed in Section A4.2-8. Consequently, the reactor shutdown heat removal will have to be accomplished by one of the two remaining diverse paths, mainly the auxiliary cooling system (ACS) or the reactor vessel air cooling RVACS.

As stated in Section A4.2-8, the two condensate pumps are each rated at 50% capacity of a power block. The feedwater pumps are rated at 50% capacity each. Hence, one condensate train and one feedwater train will be able to remove decay heat from all three modules of a power block; provided of course, that all modules are shutdown. If one module is not shutdown, then one condensate train and two feedwater trains will be needed to maintain it running at full power while removing the decay heat from the other two shutdown reactors.

X. RELEASES

AX-1 Comment

The following questions relate Accident Type F3, Core Damage Categories C6 and C6S, and Release Categories R4A and R4AM:

(a) In the "top events" of the F3 event tree and its description in the test, it is implicitly assumed that the energetics event is due to sodium voiding reactivity alone. Perhaps this would be the case if all inherent feedbacks were inoperable. Suppose, however, that the initial voiding event were not sufficiently energetic to cause a disassembly and that, alternatively, the accident entered a "transition phase." This possibility appears to have been ignored in the analysis. What are the likelihood and consequences in terms of radiological release and risk of an energetic event resulting from "conceivable" molten fuel motion recriticality events in a voided core? What are the analyses and supporting data base?

(b) The fission products released during the energetic excursion are assumed to be deposited in the Head Access Area. They are released to the atmosphere from this region as a result of leakage from the HAA. It would seem possible that the excursion would lead to failure of the HAA enclosure by, perhaps, a missile created by the event or by pressurization of the HAA. If this were the case then the inventory of fission products released during the event would be available to the atmosphere immediately, rather than by a leakage flow process. What is the analysis which leads to the assumption of integrity of the HAA during this scenario? What are the consequence on risk if "massive" failure of the HAA should occur immediately?

(c) Core Damage Category C6 results from an energetic excursion involving melting of 100% of the core and vaporization of 10% of the core. If the HAA were to fail, then the available fission products would be available for immediate release to the atmosphere. The consequences of this "early containment failure" could be severe, depending on assumptions made with respect to availability of airborne fission products at this point in the accident progression. The PRA assumes that 100% of th noble gases are deposited in the HAA along with 5% of the fuel in aerosol form. Does the "fuel" include its plutonium content?

The remaining fission products are assumed to remain with the primary sodium and they are subsequently released over a long period of time as the sodium burns as opposed to being released early (upon containment failure). This assumption requires justification in terms of a mechanistic scenario. It appears plausible that the high-pressure bubble which expands against the sodium pool would contain all the fission products generated during the excursion. The inventory of fission products released during the excursion would depend upon the temperature of the melt upon the termination of the excursion. If the temperatures are high enough then even the lower volatility fission products may be driven from the fuel into the bubble.

temperatures are high enough then even the lower volatility fission products may be driven from the fuel into the bubble. The bubble expands against the sodium and fails the head. The fission products within the bubble may end up in the HAA or in the reactor building immediately rather than being dissolved in the sodium. Describe the accident scenario, specify the melt temperature subsequent to the excursion and the fission products that are likely to have been released at this temperature. Describe the mechanisms by which the fission products would be retained in the sodium in the context of this energetic accident sequence. What would be the consequences of an alternative hypothesis which would lead to deposition of a substantially higher inventory of fission products in the HAA?

Response

(a) Although explicit reference to transition phase scenarios has not been included in the headings of the core response event trees ans in the text discussing the loss of flow accident F3, the probabilities of energetic events have accounted for such scenarios. Specifically, these probabilities were assigned as follows:

As stated in Section A4.3.4.3, Accident Energetics, "Accident analyses conducted so far for the PRISM metal core covers only slow transients where delayed feedback mechanisms become effective. More severe accidents involving, for example, coherent sodium voiding resulting from failure of pump coastdown, could lead to superprompt critical core (net reactivity greater than $1) and subsequent energetic disassembly. To assess the consequences of such accidents on radioactive material release from the core and on the structural integrity of the vessel, use has been made of the parametric evaluations in Reference 1 and reported accident analyses for the FFTF in References 2 and 3.

Reference 1 conducted parametric evaluations to study the effect of the Doppler coefficient, power flattening, and the equation of state, among other factors, on the explosive energy resulting from reactivity additions. Review of these evaluations indicated that the PRISM core and the FFTF core should have comparable energetics under severe transients leading to core disassembly. Consequently, the FFTF accident analyses reported in References 2 and 3 were used as a basis for assessing the core response for PRISM accidents involving sodium voiding."

The analyses in Reference 3 include an assessment of energetics due to early disassembly and transition phase scenarios for loss of flow accidents in the FFTF. Based on the analysis in this reference, core damage category C6, which leads to the most severe core and vessel damage, was defined. This category corresponds to Reference 3 energetic event of 20 to 35 MW seconds, which leads to vessel head seal failure, and the expulsion of 5% of the core in aerosol form into the head access area.

References:

1. "Fast Reactor Meltdown Accidents Using Bethe-Tait Analysis," R.A. Meyer and B. Wolfe, Advances in Nuclear Science and Technology, Vol. 4, 1968, p. 197.

2. "An Analysis of the Unprorated Transient Overpower Accident in the FTR," A.E. Walter, et al., HEDL-TME-75-50M HEDL, June 1975.

3. "Risk Analysis Application to FFTF - Second Cycle," K.G. Feller and D.E. Hurd, GE, GEFR-00029, January 1977.

(b) With respect to failure of the Head Access Area structure as a leakage boundary due to missiles or over-pressurization during an early energetic excursion, these possible events were judged not to have a large effect on subsequent release of fission products because the HAA was already assumed to have open leak paths. The existence of these open leak paths does results in prompt expulsion of some fission products at the beginning of the event. This can be seen from the release fraction of solids (curve E) in Figure A4.4-15 for category R4A. This early release is relatively unimportant for volatiles and noble gas because the actual volume of vapor phase expelled compared to the total volume at vapor phase or sodium phase is small. Only catastrophic failure of the HAA structure would cause a major effect and this is very unlikely since there is a leak path to vent the pressure, so a very high energetics would be required.

(c) The deposition of 5% of the fuel in the HAA during an energetic expulsion does include Plutonium. Studies of the transport of fission products within the bubbles formed during an energetic LMFBR excursion have been conducted at ANL showing deposition of fission products onto the surface at the bubble, hence into the liquid Na phase. Detailed discussion and quantification of these effects was considered beyond the scope of this study. The release fractions used were based on information supplied by ANL in ANL-PRISM-5.

AX-2 Comment

In the containment response sequences which lead to large scale core meltdown, the judgment is made that the probability of a "late energetic expulsion" is 0.01. This assumption assigns a low probability to the possibility of energetic recriticalities due to voided-core fuel motion events at this stage of the accident. Describe the phenomenology and supporting data base which justifies this level of probability.

If the recriticalities are not energetic and do not threaten the vessel, the molten fuel temperatures would be increased and a larger quantity of non-volatile fission products could be released. How would this scenario influence the source term and the associated risks?

Response

The value 0.01 was assigned as the probability of late energetic expulsion subsequent to core meltdown by experts at Argonne National Laboratory as reported in ANL-PRISM-5, page 4-65. Details of the basis for such assignment were not given but the probability of energetic re-criticality was stated to be very low since it requires large scale collapse of remaining fuel structures.

For the case where such late re-criticalities occurred but were not energetic, the source terms used in this study would not be affected since the assumption has already been made that all fuel becomes molten, given large scale core melt. Thus, the source terms used release 100% of all elements that are even close to volatile at fuel melting temperatures. Studies have shown that further release of substantial amounts of radioisotopes at temperatures beyond melting is a very weak function of temperature.

AX-3 Comment

In those accident sequences leading to early head venting by warpage or seal failure describe the calculation of flow of fission products from the vessel to the HAA to the reactor building. What are the features which limit the flow from the HAA to the surroundings?

Response

In sequences involving head venting by warpage or (R6A) seal failures, the failure is caused by temperatures and pressures resulting from heat up and boiling of Na requiring about 25 hours. In these scenarios noble gases are released from the fuel to the cover gas space and HAA at the time of core melt (R6A). Subsequent leakage from HAA is via leak paths around access hatch and other penetrations. The pressure driving this leakage flow is due to boil-off of sodium vapor venting from the reactor vessel head into the HAA. Boil-off of the remaining Na requires 25 hours, during which time most of the noble gas and about 60% of the Cs, I and Te (volatiles) leak from the HAA along with the Na vapor. The Na vapor burns as it leaks into the atmosphere forming NaO aerosol. This leaking Na vapor also carries with it a small fraction of the solid elements in the fuel in the form of aerosol-sized particles.

AX-4 Comment

Describe the basis for Release Category R3. This category appears to include a large variety of sequences, including ones with and without "late energetic expulsions," and ones with and without failure of SHRS. Why is there no distinction made between these apparently different sequences with presumably different release patterns?

Response

Certain outcomes of the containment response event trees C6 and C6S involving early energetics as well as failure of shutdown heat removal were erroneously assigned to release category R3, which has no long-term source term but only that associated with the early energetic event. This has been corrected by re-assigning these outcomes to release category R4A, the most severe release, which does have the long-term source.

AX-5 Comment

Describe the physical scenario and assumptions made with respect to fission product release for Release Categories R6A, R6U, R8A, R8U and R8S. What fuel temperatures are assumed, what fission products are released in what quantity? Describe the flow paths for the fission products from fuel release to the atmosphere taking into account whether the fission products are released into the sodium pool or into the gas region of the primary vessel.

Response

The six release categories with prefixes R6 and R8 all represent essentially a loss of shutdown heat removal scenario. The various events in addition to the loss of SHR have only a second order effect on public health consequences. The sequence of events involving fission product transport are essentially these described under the response to Comment VIII-. For this event, sodium bonded metal fuel melts at 1150°C, the Iodine isotopes are held as CsI in the fuel (boiling point 1280°C) or as NaI in the sodium bond (b.p. 1304 °C). Hence, the iodine does not enter the gas phase upon fuel melting, because the boiling points of the iodide components have not been reached. Thus, Iodine enters the liquid phase sodium. This Iodine is the dominant isotope as far as early health effects. Although some Cs may enter the gas phase, the resulting health effects will be the same assuming that it follows the liquid Na as it boils off because Cs is a minor contributor to early health effects. Cs does affect long-term health effects but timing is not so important as total fraction released. One hundred percent of Cs, Xe, Kr, and I are assumed released from fuel. Some of the Cs and I remain within the HAA due to aerosol settling and plate-out during the process.

AX-6 Comment

In sequences leading to deposition of debris within the concrete silo, decay heating and chemical reaction energy will eventually lead to sodium depletion (after several days). What is the temperature history of the debris following sodium depletion? What fission products are released to the environment? Was this release included in the consequences results presented in Tables A4.5-1 and A4.5-2?

Response

The long-term release of solid aerosol from the core debris following sodium dryout was not addressed due to the limited scope of this PRA and the early conceptual nature of the design. The judgment was made that such a release would be a minor contributor to health effects due to several factors. Since water has been depleted from the concrete, the driving pressure force and material flow involved in entraining aerosol is much weaker than in the earlier stages with Na-water reactions present. In most melt-through scenarios shutdown heat removal has been lost, often due to severe blockage of RVACS passages by debris. This debris combined with the large surface area of the RVACS exhaust passage would cause plate-out of most of any aerosol generated. Finally, the time scale for such release has been shown to be of the order of a month. In this time frame, emergency response by institutions will likely involve limiting any such long-term release process.

XI. MISCELLANEOUS

AXI-1 Comment

Please provide the fault tree analysis that was done to assess the frequency of initiating events 1, 2 and 3.

Response

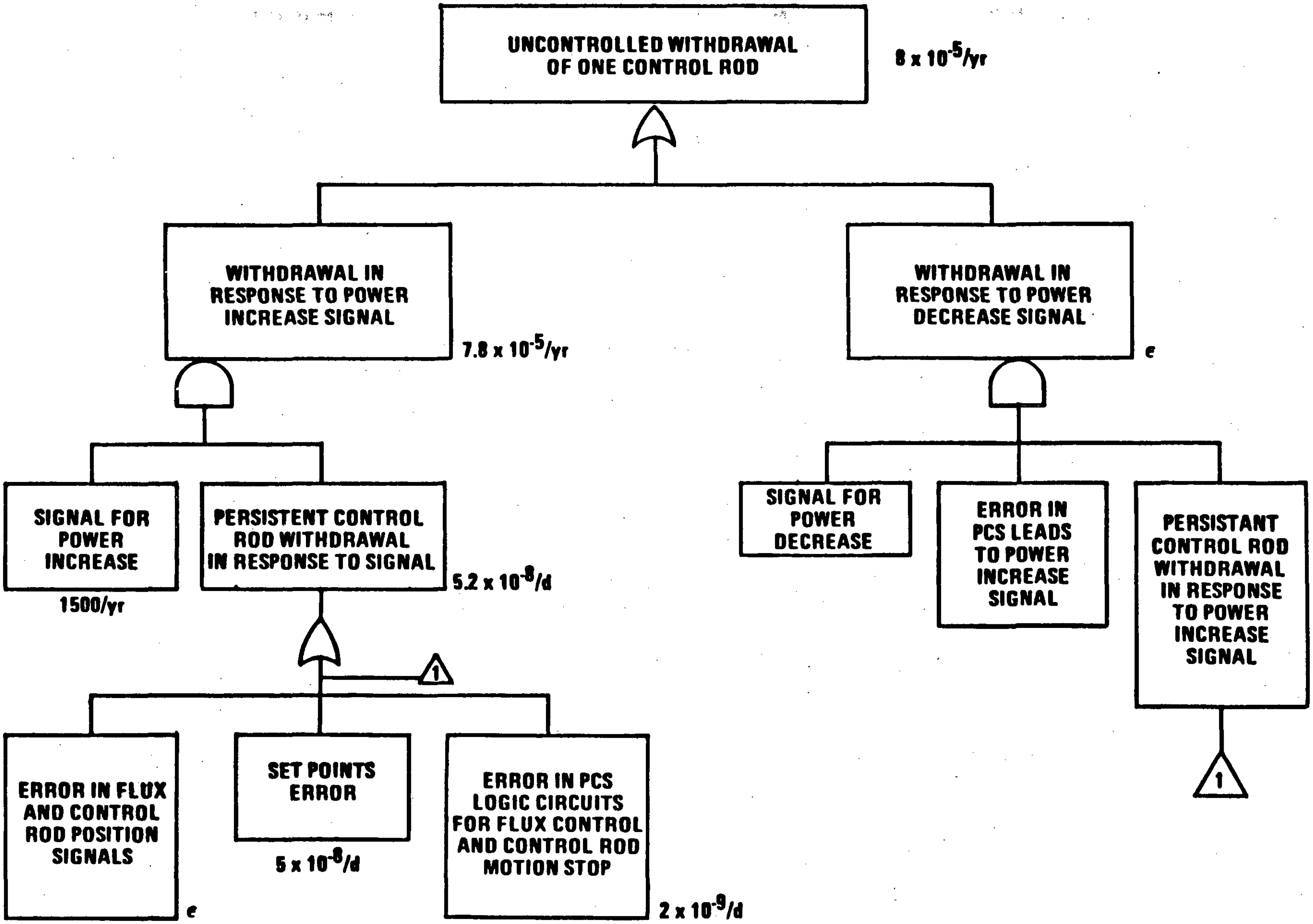
The fault trees for initiating event 1 is enclosed. As stated in Section A4.1, this tree is a preliminary functional fault tree. At the first level of the tree, the top event is divided into events initiated by power increase demand and the complement event of power decrease demand. This level is divided into the necessary conditions that 1) a signal is received by the PCS flux controllers to change power, and 2) the PCS withdraws the rod in response and continues withdrawal until the rod is completely withrawn. This latter event is then divided at the third level to 1) erroneous flux and control rod position sensors signals, 2) errors in the PCS setpoints, and 3) failure in the logic of the PCS which leads to continued withdrawal of the rod.

The fault tree shows a frequency of power increase requests of 1500/year which conservatively accounts for power load following needs. The probabilities of setpoint errors and PCS logic failure have been obtained from the PCS analysis of Appendix A, Section A4.2.

As seen on the fault tree, frequency of the top event is estimated as $8x10^{-5}$ per year. A frequency of $1x10^{-4}$ per year has been used in the PRA of Appendix A.

As stated in Section A4.1, the frequency for initiating event 2, control withdrawal, has been conservatively assumed to be the same as that of IE-1, i.e., 10^{-4} per year.

For initiating event 3, credit was taken for the fact that more failures must occur for reactivity additions more than those of IE-2. This credit was subjectively assignd as one order of magnitude reduction in the frequency of occurrence of IE-2. Thus the frequency of IE-3 was assigned the value $1x10^{-5}$ per year in the PRA.



FUNCTIONAL FAULT TREE FOR CONTROL ROD WITHDRAWAL

87-580-02

AXI-2 Comment

How do you estimate the frequency of IE16? How does the MTTR for IE16 compare with that experienced by LWRs? Does IE16 cause EM pumps to trip? Does tripping an EM pump lead to an automatic reactor trip? Why is IE16 said to be not an initiating event (see Page A4-11). Are operator actions needed to mitigate the consequence of IE16? How does IE16 affect the pony motor of the intermediate sodium pump?

Response

Initiating event IE16 is defined as loss of the capability to provide electric power sufficient to remove the operating power heat load. This means loss of all off-site and on-site electric power sources capable of running the BOP, IHTS and primary pumps. The frequency of occurrence of the event is obtained from data on off-site power outage frequencies and duration and on-site power reliability and maintainability estimates. The frequency and MTTR for the event were obtained as follows. The failure rates for off-site power and on-site power from one power block are 10^{-5}/hour, and 10^{-4}/hour, respectively, with associated repair times of 1/2 hour for off-site and 1000 hours for on-site. It is assumed here that power can be supplied to the primary pumps from either off-site power or from either of two power blocks on-site. Thus the frequency of loss of all power is calculated as follows:

	Frequency	Repair time	Unavailability
Off-site	10^{-5}/hr.	0.5 hrs.	5×10^{-6}
On-site block 1	10^{-4}	1000	0.1
On-site block 2	10^{-4}	1000	0.1

Loss of all three power sources:

$$\text{Unavailability } Q = 5\times10^{-8}$$

$$\text{Residence time } T = (1/t_1+1/t_2+1/t_3)^{-1} = 0.5 \text{ hour}$$

$$\text{Frequency } F = Q/T = 10^{-7}/\text{hr} = 8\times10^{-4}/\text{year}$$

The scenario for such an event is that first one onsite block becomes unavailable, then, while it is under repair, the second block fails. The residence time for both blocks in a failed state, each with a 1000 hour repair time, is $(1/1000 \text{ hr} + 1/1000 \text{ hr})^{-1}$ = 500 hours. Then, during this 500 hours, off-site power loss occurs. However, this scenario will not occur because there will be safety related technical specification that the plant not operate for more that some period (say, 36 hours) with both on-site power sources down. Hence, the above calculation of frequency must be reduced by a factor of 36/1000. The resulting frequency of loss of all power to a module while the module is operating is 3×10^{-5}/year.

A repair time of 1200 hours is conservatively used for this event based on the assumption that the affected module will be subjected to an inspection following such a transient.

The station blackout IE-16 will lead to a loss of electric power to the EM pumps. The pumps will coastdown using the 1E-grade coastdown synchronous machine. The reduced flow will cause a reactor trip. Tripping even one pump will cause an automatic reactor trip. No operator action is involved in the above sequence. On loss of all power, the on-site auxiliary gas turbine generators will start automatically and will deliver power to the IHTS pony motors. Once one module starts operation, it can deliver power to all IHTS pony motors of the nine modules in the reference PRISM plant.

AXI-3 Comment

Page A4-46 - Is case 1 conditional on failure of the operating power heat removal function? Does failure of block 130, e.g., pony motor, cause failure of both the operating power heat removal and the decay heat removal via IHTS? Please provide fault trees for these top events in the system event trees.

Response

Case 1 p. A4-46, has an error in its definition and should be corrected to read:

> "Conditional probability that decay heat removal via IHTS and BOP fails given the initiating event."

Also, the second heading in Table A4.2-15 should be corrected to read:

> "Via IHTS and BOP Given IE"

The third heading in the Table should be corrected to read:

> "Via RVACS given failure via IHTS and BOP and given IE"

These corrections are included in Amendment 8 issued November 20, 1987.

In response to the second question, failure of block 130 does not necessarily mean that the operating power heat removal has also failed. In particular, failure of the pony motor does not affect the IHTS flow during power operation. In the power operating mode, the IHTS is driven by the main motor. On the other hand, structural failure in the IHTS will affect heat removal during power and shutdown modes. Therefore, and as shown in Table A4.3-13, initiating events which include such structural failures, e.g., IE 14, fail block 130 with a probability of 1.0.

Fault trees which have been developed for the top events in the system event trees are contained in Section A4.2 of the reactor protection system, reactor shutdown system, and pump coastdown system. Fault trees for other systems in the system event tree have not been developed and are not available at this time.

AXI-4 Comment

Referring to page A4-37 - What are the loss of coolant accident initiators considered? What will happen if a pipe break occurs in the pump discharge line or return line of the primary sodium purification system?

Response

The loss of coolant accident initiator referred to in page A4-37 is the local core blockage initiating event, IE-8. The following correction will be made in the related text: "To evaluate the common cause failure probability, a beta factor of .05 is conservatively assumed for the reactivity insertion and local core blockage initiating events." See Amendment 8.

Results of a scoping study to evaluate the effects of postulated leaks in a primary pump discharge pipe without scram are summarized in Table 1. The "large" leak size corresponds to a hypothetical double-ended guillotine break at the approximate pipe mid-point. The "small" leak corresponds to a flow area of the order of 1 inch^2. It is noted from the table that flow through the core would be about 43 percent of normal operating flow rate under the worst condition, i.e., a double-ended guillotine pipe break. If such an event were to occur the core exit temperature would increase and the pump head would decrease as indicated. However, the RPS will trip the reactor when the power/flow setpoint or core exit temperature setpoint is reached, whichever is reached first.

Table 1
Operating Parameter Summary
With Primary Downcomer Pipe Break/Leak

Operating Parameters	No Leak	Leak Size: Small	Medium	Large
Flow Split (Core, %/Break, %)	100/~0	91/9	75/25	43/58
Pump Head (psi)	120	102	68	21
Core Exit Temperature (°F)	785	893	952	1171

As to the return line of the primary sodium purification system, this line is isolated with safety class double valves during operation. Sodium purification of PRISM is done only during shutdown.

AXI-5 Comment

Many hurricanes exceed 130 mph wind speeds, although PRISM's "safety related" structures are designed to withstand up to 130 mph winds. It has also been states that (with the exception of tornadoes) meteorological extremes will have no effect on the continued routine operation of the plant. Is this true for extreme hurricane wind speeds?

Response

It is recognized that hurricane wind velocities can exceed 130 mph. Safety related structures are designed for Region I tornado wind velocity of 360 mph combined with missile impact loads which are expected to envelope hurricane wind effects. These structures are designed to protect safety related equipment and systems to ensure safe plant shutdown in the event of either tornadoes or hurricane extreme wind effects. Should hurricane wind velocities exceed 130 mph, routine operations of the plant may be interrupted pending evaluation of wind effects on the structure.

AXI-6 Comment

Why has a dropped fuel assembly during refueling not been considered an initiating event?

Response

A dropped fuel assembly event has been considered during the design of the handling equipment for the reactor refueling outage.

a) At the reactor, the In-vessel Transfer Machine (IVTM) has features which minimize the possibility of dropping a core assembly. The grapple head is designed with two fingers that are actuated by a cam to extend outward into two holes located in the core assembly handling socket. The clearances between the socket and the grapple head are sized such that, in the event one of the fingers fails, the assembly will remain attached by the remaining finger, until it is transferred to a storage location and the fingers are retracted into the grapple head. Failure of any other components in the grapple finger actuation drive will not result in a dropped core assembly.

An interlock is provided to prevent unlatching a loaded grapple at an elevation which does not correspond to the core, the storage position or the transfer station. An additional interlock is provided which prevents unlatching the grapple with a load on it. These two interlocks dictate that the grapple is at the correct elevation and the core assembly is fully supported prior to unlatching the grapple fingers.

For operation of the IVTM, procedures will be followed that would minimize damage caused by a dropped core assembly. During a transfer, the grappled assembly is no more than 6 inches above the tops of the core. When the IVTM is directly in front of the storage positions or the transfer station, the assembly is then raised to the elevation required to complete the transfer.

In the unlikely event that the IVTM drops a core assembly it will be necessary to manually grapple the assembly and raise it into a cask. The access hatches, located in the rotatable plug, will be used to retrieve and remove the assembly. Due to the narrow confines of the reactor vessel, the dropped assembly will maintain an upright orientation with the bottom supported by the top of the core and the top of the assembly leaning against the support skirt or the Upper Internal Structure. The position of the assembly will permit manual recovery by grappling of the handling socket and removal to a safe location.

b) At the Fuel Transfer Cask (FTC), the fuel assembly is lifted from the reactor after being placed into a basket (pot). The basket is mechanically attached to a bi-stem (double hoisting cable) that lifts both the basket and fuel assembly into the FTC for transfer to the Fuel Service Area (FSA). By placing the fuel assembly into another component which has a redundant cable lifting system and no grappling type action for attachment, the dropping of a fuel assembly can not occur during this phase of fuel handling.

c) At the Fuel Handling Cell (FHC) in the FSA, a dropped fuel assembly would be detected by the operator who is carrying out the transfer and a second operator who is always present when irradiated core assemblies are handled. The FHC grapples have a positive-locking finger design to prevent the grapple fingers from inadvertently retracting much the same way described in a). The FHC floor is covered by a grid to prevent contact of a fallen assembly with the floor liner, thereby avoiding hotspots and potential failure of the liner if this event would happen.

The FHC will also be equipped with a core assembly pickup tool that is used with the powered manipulator to retrieve and restore the assembly to a safe location.

XII ADDITIONAL QUESTIONS ON THE ANL METAL FUEL PROGRAM

The following responses to the NRC comments related to the ANL Metal Fuel Programs were provided to GE by ANL. See page F4-64 for additional discussion of the Metal Fuel Program.

AXII-ANL-1 **Comment**

The neutronic event which would result from ULOF with failure of inherent feedback would begin with sodium boiling and voiding the coolant channels in the high power regions of the core. This positive reactivity insertion would be accompanied by a rapid power rise which would lead to rapid fuel heatup, melting and fuel element failure. If the fuel fails near the core midplane this would lead to an additional positive reactivity insertion, which could only be immediately mitigated by sweepout in the coolant channels. The "follow-on", low-power channels would, it would appear, also be likely to experience near-midplane failures. In the light of this scenario, justify the contention that this class of accident would lead to "low energetics." What data base is relevant to prediction of fuel failure prediction under these conditions?

Response

See response to AXII-ANL-2

AXII - ANL-2 **Comment**

Energetic "transition phase" events are considered of low probability such that, for example, the probability of Release Category 6A is at least two orders-of-magnitude lower than that of Release Category 8A. Considering that at the fuel melting temperature stainless steel is unavailable as a "dispersal" driving force and that bond sodium may be displaced to the fission gas plenum it appears that fuel compaction could occur. What is the justification for the belief that energetic events are unlikely?

Response

The ANL-suggested source terms for "energetics scenarios" for consideration in innovative design PRA's have been reduced from a level characterized by actinide inventories of 10% of core inventory to a level with 1% core inventory [1]. The original 10% was chosen somewhat arbitrarily for use in ANL PRA's [2,3] because it represented an enveloping value used in earlier bounding assessments [4] of oxide LMLR concepts such as LSPB. The reduction to 1% was deemed appropriate because of a number of qualitative considerations relating to the metal fuel system (vs. the oxide system) and design characteristics of the innovative concepts.

In theory, energetics involving vaporization of significant fuel inventory (1-10%) in fast reactor accident evaluations can be postulated as the result of several mechanisms inducing large and rapid reactivity excursions:

(1) Rapid rod runout or expulsion

(2) Coherent and autocatalytic mid-plane failure with coincident rapid voiding events

(3) Recompaction

(1) The design philosophy for the innovative concepts has been to lower the rod worths to levels on the order of several cents and to recommend Tech Spec limitations on the testing tolerances of total rod bank reactivity worth. Mechanical limitations on rod withdrawal speed are also cited. Thus rapid, high worth runouts capable of fuel-vaporizing reactivity excursions are not an issue for the innovative designs.

(2) There is a clear propensity for top-of-pin failures evidenced by the ongoing M-Series of experiments [5] simulating TOP scenarios taken to core disruption. A bias toward top-of-pin failure location for metal fuel pins relative to oxide pins is, of course, expected due to the shorter time constants in fuel-coolant heat transfer. The high thermal conductivity of metal fuel assures peak cladding temperatures and therefore preferred failure location near the top of the pin for plausible TOP scenarios (failure in the M-Series occurred near four times nominal power). For LOF scenarios, there is an analogous expectation that pin failure would occur higher in a metal fuel core than they would in an oxide core. However, TREAT experiments simulating LOF's to core disruption in metal cores have not yet been undertaken.

Post-failure fuel dispersal is more favorable for metal fuels. In the M-Series when cladding failed, post-failure events were characterized by rapid fuel dispersal, rapid - but temporary - coolant voiding, and partial flow blockage exhibiting little potential for subassembly blockage. Different fuel types tested behaved similarly. Pressure spikes were minor (less than 1 MPa) and were correlated to the plenum pressure of the failed pin (but about an order of magnitude lower). In each case about half of the fuel inventory was ejected through a small beach at the fuel top. On a qualitative basis, post-pin-failure material motion was benignly dispersive in all tests over a range of metal fuel types.

(3) Transition phase scenarios involving rapid recompactions are hard to conceive with the metal core. First, with the extended time frame of metal fuel scenarios in the innovative designs, the absence of a mechanistic, coherently acting pressure source is even more obvious than it was for the case of an oxide core. Nest there is the demonstrated propensity for eutectic formation between the fuel and steel and the resulting effect on a transition phase geometry. The fact that eutectic formation temperatures for the range of fuel compositions and clad types are similar to the temperatures typical of core region sodium during accidents suggests extensive fuel relocation before blockages would form. The M-Series tests cited above provide evidence of significant intrasubassembly fuel relocation. Downward melt relocation is being specifically investigated as part of the out-of-pile debris testing.

Out-of-pile unirradiated fuel debris coolability tests continue to show far higher porosities than those typically expected for oxide debris beds. Porosities of the order of 90% have been observed in injection tests [6] wherein kilogram quantities of fuel were injected into a pool of sodium. Criticality calculations of 90% porous fuel configurations with varying compositions and geometries show strong subcriticality. Thus, such highly dilute fuel inventories would obviously work against the possibility of reconfiguration into supercritical masses through recompaction. As a result, in-core debris coolability assuring permanent subcriticality prior to whole core involvement in unprotected scenarios is a strong possibility, and research is oriented to gaining more knowledge of the underlying mechanisms.

The above evidence suggests that the likelihood of fuel motion and voiding sufficient to yield high ramp rates is much less for the metal-fueled innovative designs than for the oxide cores on which the original source terms were based. Even if a reactivity excursion occurs, it would be expected to vaporize less fuel than an equivalent excursion in an oxide core because the initial fuel temperatures would be much lower and the boiling temperatures higher. With the very rough but still representative values or fuel properties shown below, it is clear that the energy expended per unit fuel mass in an

	Metal	Oxide
Melting Temperatures (°C)	1100	2800
Vaporization temperatures (°C)	4100*	3300
Specific heat (j/g-°C)	0.2	0.4
$C_p(T_v - T_m)$ (j/g)	600	200
Full density (g/cc)	16	10

excursion prior to incipient vaporization is on the order of three times greater for the metal core. The volumetric energy density requirements are greater yet.

While this simplistic approach does not consider fuel dispersal dynamics, it does suggest that actinide vapor source terms would be much less in a metal (vs. oxide) core for a given excursion. In studies of core disruption in oxide cores it has been assumed that steel vaporization would provide a dispersive mechanism to prevent energetic recriticalities. However, Fauske [8] has suggested that sodium vaporization should be as or more effective in providing a means of core dispersion for metal fuels with very low energetics.

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* Obtained as an approximate solution to

$$\log_{10} (\text{vapor pressure (atm)}) = -25{,}230/T(K) = 5.71,$$

a relationship for uranium metal provided by Leibowitz (7).

References

1. C. Mueller, Presentations to the Nuclear Regulatory Commission on the SAFR & PRISM PSIDs, November 4, & 5, 1987.

2. C. Mueller, et al., A Risk Evaluation of Severe Accident Progression in SAFR, ANL-SAFR-4, January 1986.

3. C. Mueller, et al., A Risk Evaluation of Severe Accident Progression in PRISM, ANL-PRISM-5, January 1986.

4. Reactor Analysis and Safety Division, unpublished work on LPP and LSPB.

5. T. H. Bauer, et al., Behavior of Metallic Fuel in TREAT Transient Overpower Tests, Proceedings of Safety of Next Generation Power Reactor, May 1988.

6. J. D. Gabor, et al., "Characterization of Metal Fuel Fragmentation", ANL-IFR-52, August 1986.

7. L. Leibowitz, private communication, January 12, 1988.

8. C. E. Till, Y. I. Chang, J. H. Kittel, H. K. Fauske, M. J. Lineberry, M. G. Stevenson, P. I. Amundson, and K. D. Dance, "Fast Breeder Reactor Studies", ANL-80-40, July 1980.

AXII-ANL-3 Comment

The analyses of the PLOHS accidents indicate that no releases occur until sodium is boiled off to uncover the core. Sodium reaches its saturation temperature at about 25-30 hours and core uncovery begins at about 100 hours. Releases do not begin until core uncovery. This sequence assumes that the core is coolable under natural convection boiling conditions. What is your assessment of the coolability of the core under these conditions? If core disruption were assumed to begin at around 25 hours, what effect would this have on the consequences? Would this affect your assumptions about the need for off-site evacuation plans?

Response

The assumption used to develop the PLOHS scenarios proceeding to core boiling needs to be revised to account for wide-scale eutectic formation, cladding penetration, and debris formation. Since eutectic formation is very rapid at temperatures typical of sodium boiling under accident conditions, the core would disrupt long before it was uncovered. Although coolability would be expected, release of the volatiles would begin upon head leakage.

Nonenergetic events contribute primarily to risk through latent effects. Thus evacuation times should be insensitive to leak initiation times, other things being equal.

AXII - ANL-4 Comment

What is the assumed chemical form of iodine released from fuel upon fuel element failure? What is the relevant data base?

Response

In metallic fuels the iodine will form iodides with cesium and sodium. Because of the large volume of primary and bond sodium in the core, most of the iodine will be in the form of NaI dissolved in sodium while the cesium will be in solution in the fuel alloy and in the sodium.

Cesium and iodine releases to the bond sodium have been measured in EBR-II Mark-II fuel by Villarreal. [R. Villarreal, "Distribution of Fission Products Released from Breached Mixed-oxide, -carbide, -nitride, and Metallic Fuels Irradiated in EBR-II', Proc. Int. Conf. on Fast Breeder Reactor Fuel Performance, Monterey, California, March 5-8, 1979, p. 585 (1979).] Gamma scan analyses of cesium distributions [B. R. Seidel, et al., ANL-RDP-93, p. 6.19 (March 1980)] have confirmed Villarreal's chemical analyses. Recent chemical analyses of a 5.3 a/o burnup U-10Zr pin and 2 a/o burnup U-19-Pu-10Zr pin show similar cesium releases to the bond sodium [R. Villarreal, Private communication] as those measured in EBR-II Mark-II U-Fs fuel. Iodine was not measured in these experiments because most of it would have decayed before the measurements were taken. However, the formation of NaI is favored thermodynamically. Also, the zero detected release of iodine from breached pins to the cover gas in EBR-II is strong evidence that the iodine reacts with the sodium to form a less volatile iodide.

AXII - ANL-5 Comment

What fraction of the inventory of volatiles are found in the fuel element fission gas plenum at the time of fuel failure? How is this reflected in the proposed source terms for the various release categories? What is the relevant data base?

Response

The data in the references given in the response to Comment AXII - ANL-4 suggests that 20 to 30% of the total cesium inventory may be dissolved in the bond sodium in the plenum. Similar fractions of other volatiles may be expected because their release paths from the fuel and transport to the plenum through the interconnected porosity are similar.

The attached memo by Chasanov gives the rationale that was used in determining the source terms for the various release categories. In reviewing this memo it appears that Chasanov assumed the release of cesium from breached fuel occurred primarily from the flowing sodium leaching the cesium from C_sI that "forms in the cooler portions of the pin". He estimated that 10% of the total cesium content and 100% of the iodine and other halogens would be released from breached, but un-fragmented, fuel. Based on the above data for cesium in the bond sodium, Chasanov's estimates for cesium release are too low but the halogen releases may be too high. The subject is under current review.

In the case of severe fuel disruption where the fuel fragments, Chasanov argues that all of the halogens and all the volatile metals are released to the sodium. This is not unreasonable since fragmentation of metallic fuel requires melting; the combined high vapor pressure plus the increase in exposed surface area of the fragments would release the volatile fission products to the primary sodium.

AXII - ANL-6 <u>Comment</u>

What is your assessment of the potential for energetic molten fuel-coolant interactions (vapor explosions) during "transition phase" sequences, assuming that recriticalities could raise the molten core temperature well above the fuel melting temperature?

<u>Response</u>

Fuel-coolant interaction refers to a pressure generation event that may occur when hot molten fuel comes into contact with the sodium coolant. Such an event, which is caused by rapid transfer of thermal energy from the fuel to the coolant, is a source of mechanical energy release. The interaction can be mild or energetic, depending upon the extent of conversion of the fuel thermal energy into mechanical work. When the extent of energy conversion is significant, the interaction is called "energetic". It is this energetic interaction that is of concern in safety considerations.

While little data exists for metal fuels, the current understanding of fuel-coolant interaction phenomena indicates that the fuel temperature would have to exceed approximately 3000 K in order to meet the necessary requirements for an energetic interaction. This temperature requirement, however, is not a sufficient condition for an energetic interaction. Past work has established that an energetic interaction proceeds through several stages, namely, coarse premixing, triggering, fragmentation and propagation. Failure to achieve any one of the stages would render the interaction nonenergetic or benign. Evaluation of the various stages involved would require consideration of specific accident situations. It is generally believed that, even if the temperature condition is satisfied, an energetic interaction would be highly unlikely unless molten fuel is mixed with an optimum volume of the coolant. Sodium will have vacated the core region where such interactions would occur. Further early dispersal and the absence of plausible high ramp rate mechanisms are expected to result in maximum fuel temperatures well below the minimum temperature of 3000 K required for energetic fuel coolant interactions.

A final question concerning core disruption that must be addressed is the ultimate coolability of core debris. The coolability of a metallic fueled reactor following a core melt accident is enhanced relative to an oxide fueled reactor. The quenching of molten oxide in sodium results in very fine particles with an average diameter ~400 um with a low porosity. The limited available experimental evidence with gram quantities of material indicates that upon quenching of molten metallic uranium in sodium much larger particles are generated with a greater porosity. This enlarged particle size and increased porosity are expected to increase the coolability both by natural convection and by sodium boiling.

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INTRA-LABORATORY MEMO

April 30, 1985

To: B. W. Spencer RAS

From: M. G. Chasanov MGC RAS

Subject: Fission-Product Releases from Sodium-Cooled Metallic Fuel Elements

The renewed interest in metallic fuels for sodium-cooled fast reactors prompted this survey of available literature dealing with fission-product releases to sodium from metallic fuel elements. The object of the survey was to evaluate the extent of such releases and to determine the form of the fission products in the sodium. While the primary interest was in IFR-type fuels, the literature search covered all metallic uranium-based fuels, including those used in EBR-II and DFR.

I. Literature Study

The assistance of John Frazier (TIS) was utilized to search the various DOE energy-related data bases for publications dealing with fission-product releases for alloy nuclear fuels. There were 572 citations listed in the computer outputs, but, of these, only a handful dealt with releases other than noble gases. Winnowing through those reduced the germane references still further. Apparently, in the period during which the majority of the work on metallic fuels was done, the late 1950's through the 1960's, the indexing of publications by keyword was not as detailed as is the norm currently. Thus, there may be some pertinent references that are irretrievable by this method of recall. A manual search of Nuclear Science Abstracts covering the period of 1957 through 1971 produced 32 additional citations. Again, few of these proved to be pertinent to the objectives of the search; much of the information in these references dealt with releases of noble gases from non-metallic fuel elements.

An experimental study by Davies and Drummond (1) on the behavior of fission products in DFR liquid-metal coolant concludes that the main oxide-forming fission products are present in the NaK as a suspension of insoluble particles while elements that form less stable oxide than sodium oxide are present in elemental form or in chemical combinations.

Clough (2) found that the behavior of barium and strontium in reactor-grade sodium, both in transport and vaporization, indicates their presence as involatile oxides.

Erdman et. al. (3) summarizing the work up to 1973 conclude that:

(a) The noble gases escape from the sodium to the cover gas in minutes,

(b) iodine is present in the sodium as dissolved sodium iodide,

(c) cesium is in elemental form in the sodium,

(d) the low volatibility of alkaline earth metals, e.g. Ba and Sr, in sodium indicate their presence as some sort of oxide compounds,

(e) the rare earths, e.g. Ce, La, etc., are not found to any significant extent in the sodium,

(g) no information is available for transition metal retention in sodium, e.g. Zr, etc.

For these conclusions, Erdman et al. relied heavily on the work of Castleman and Tang (4) and Plumlee and Novak (5), among others.

The extent of release of fission products and fuel from metallic fuel elements is not given in the citations covered except for a few DFR references, which are not really germane. A test of an EBR-II driver fuel element (7 at.% burnup) that had about one inch of fuel pin exposed to flowing sodium for four full-power days revealed no loss of fuel and only recoil release of delayed neutron precursors (6). Therefore, at this moment, there appears to be no experimental guidance for estimating release fractions from metallic fuels leached by sodium.

II. Fuel and Fission-Product Release Fraction Estimates

The extent of fuel and fission-product release to sodium by fuel elements that are breached or fragmented depends on the location of the fission product in the fuel, the form of the fission product in the fuel, its rate of release from fuel, and its interaction with sodium. In the case of a breached fuel pin, the rate of sodium exchange between the pin and the coolant may be the most important determinant of fission-product release; for a fragmented fuel, the degree of break-up may be the most important aspect. Thus, at this stage of development of the IFR concept, only general estimated guidelines for fission-product and fuel releases are feasible. When experimental data become available for IFR fuels, more detail concerning such releases will be possible. For the time being, we shall modify the estimated release fractions for oxide fuels to provide our metal fuel estimate.

(a) Breached Fuel

In the event of breached fuel we may assume the following releases from the fuel element:

(1.) All the noble gases are released,

(2.) all halogens are released,

(3) about 10% of the cesium is released,

(4.) the remaining fission products and fuel remain in the fuel element.

The rationale for these assumptions is based on the assumed transport of Cs and I to the cooler region of the fuel where access to flowing sodium via the breach is greater. It is assumed that the CsI in the pin is released into the coolant as metallic cesium and as NaI, both dissolved in the sodiuum. The 10% release for the Cs is based on the assumption that the corresponding amount of CsI forms in the cooler portions of the fuel pin. The remaining fission products and fuel are assumed to remain essentially unreleased by the breach in the fuel element. In fact, the cesium and halogen releases are probably conservative in view of the EBR-II driver fuel experience (6) mentioned above. Long term exposure, as in the case of DFR vented fuel elements, may indeed result in release of other fission products and fuel, but these releases appear to be small compared to those for iodine and cesium (1).

Fragmented Fuel

Fuel fragmentation allows access of the entire fuel cross section for release of material to sodium rather than merely the outer periphery, as is the case for breached fuel elements. If we assume that the fragmented fuel is finely divided, so that leaching of materials from the fuel takes place quickly, the extent of release would be determined primarily by the solubility of the leached material in the sodium. The solubility, of course, depends on the nature of the solid phase in equilibrium with the solution, be it element or compound. In addition, even though a fission product is essentially insoluble in the sodium, it may nevertheless be held in suspension in the coolant as fine particulate matter.

Consideration of sodium solubility data (7) and the limited experimental data lead to the following generalizations for the release fractions from fragmented fuel:

(1) All the noble gases are released to the cover gas,

(2) all the halogens are released to the sodium,

(3) all the volatile metals are released to the sodium, e.g. Cs, Rb, Te,

(4) the rare earths, e.g. Ce, La, etc., are released to the sodium to only a minor extent, probably forming insoluble oxide with oxygen in the sodium,

(5) the alkaline earths, e.g. Ba and Sr are probably released to the sodium, but form insoluble oxides,

(6) the noble metals, e.g. Ru, Rh, Pd, etc., are released to only a minor extent,

(7) the transition metals, e.g. Zr, Nb, Mo, etc., are released to only a minor extent; they may or may not form insoluble oxides depending on the oxygen level in the sodium,

(8) fuel is not signicantly released to sodium.

In short, the release fractions, from fragmented fuel to sodium, appear to be essentially either zero or unity. However, if fragmented fuel is suspended in the coolant, we can treat that as release to the sodium; therefore, even with the estimated release fractions of zero or unity, effective release fractions will vary depending on the extent of fragmentation of the fuel and the fraction then suspended in the sodium.

Admittedly, because of the lack of experimental data, these estimates are based chiefly on engineering intuition; yet they should be of value in consequence analysis if used cautiously. After their release to sodium, the ultimate fate of the fission products and fuel depends on the scenario of subsequent events. Whether they escape from containment or plate-out on reactor internals is of utmost importance, but that depends on events subsequent to the release from the fuel.

Chemical Form of Fission Products and Fuel In Sodium

The noble gases are virtually insoluble in liquid sodium and, hence, are not of concern here.

Iodine, which is the halogen of major interest among the fission products, will, in the presence of liquid sodium, form sodium iodide (8), which is dissolved in the sodium (at the concentration levels expected in a reactor).

Alkali metals, e.g. cesium and rubidium, in sodium are present in the elemental state rather than in combination other fission products. The volatility of these elements is greater than that of sodium and hence significant amounts could be transported from solution to the vapor phase (8).

Alkaline-earth elements, e.g. barium and strontium, are probably present in sodium as suspended non-volatile oxygen-containing species (2).

The rare-earth elements, e.g. Ce, La, etc. also form oxides more stable than sodium oxide and probably are present as suspended oxygen-containing materials.

Tellurium and antimony are soluble in sodium and the volatility of tellurium could result in some of it being partitioned to the vapor phase (8).

The noble metals and transition metals are relatively insoluble in sodium and thus could be present in sodium in extremely small amounts as elements (noble metals) or oxides (transition metals).

Fissile material is also essentially insoluble in sodium.

Therefore, with the exception of those materials soluble in sodium (e.g. Cs, NaI, Te, Sb, etc.), fuel and fission products, when present in sodium are probably there as suspended fine-particulate matter that is subject to deposition someplace within the reactor system.

IIV. Summary

A search of the literature produced few references dealing with fission-product release fractions from alloy fuels in contact with sodium. This result is probably an artifact due to the indexing system rather than a lack of pertinent work. Nevertheless, general estimates are provided here for release fractions from IFR-type fuel to sodium; these are based on the predicted behavior of the some fission products from oxide fuels. The predicted differences are small, as can be seen in the following:

Category	Oxide Fuel	IFR-Type Fuel
Noble Gases	Released to cover gas	Released to cover gas
Halogens	Released to sodium	Released to sodium
Volatiles	Released to sodium	Released to sodium
Rare Earths	Remain in fuel	Remain chiefly in fuel
Alkaline Earths	Insoluble oxide phases in fuel	Oxygen - containing phase insoluble in Na
Noble Metals	Insoluble in sodium	Indoluble in sodium

The release fractions for a specific accident can be estimated using the above predicted releases in conjunction with the scenario. Once the sodium starts to volatilize, the fraction of the release of halogens and other volatiles from the coolant can be evaluated using Castleman's relations (8).

MGC:ljm

Attachment

cc: J. Marchaterre
C. Mueller
A. B. Rothman
D. R. Pedersen
RAS Files: A15
EMS Files: Chron

References

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3. C. A. Erdman, J. L. Kelly, M. Kirbiyik, and A. B. Reynolds, "Radionuclide Production, Transport, and Release from Liquid-Metal-Cooled Fast Breeder Reactors", Report No. NE-4146-102-73, Sept. 1973.

4. A. W. Castleman, Jr. and J. N. Tang, "Fission Product Vaporization from Sodium Systems", in ANL-7520, I, p.540, Nov. 7-9, 1968.

5. D. E. Plumlee and P. E. Novak, "Measured Loss of Selected Fission Products From High-Burnup Mixed-Oxide Fuel in a Miniature Pumped Sodium Loop", GEAP-13731, August 1981.

6. "Status of Performance and Fabrication of Metallic U-Pu-Zr Fuel for the Integral Fast Reactor", Argonne National Laboratory, Idaho Falls, July 1984.

7. M. Hansen and K. Anderko, Constitution of Binary Alloys, Mc-Graw-Hill Book Company, Inc., New York, 1958.

8. A. W. Castleman, Jr., Nuclear Safety 11, 379 (1970).

B

C

D

E

RESPONSES TO NRC COMMENTS
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E.1 Comment

Figure E.1-1 shows the calculated reactivity changes due to bowing as a function of power to flow ratio, as calculated using NUBOW-3D. This analysis is clearly static, i.e., for equilibrium condition. How is the transient behavior factored into ARIES?

Response

At each time step during the transient analysis, ARIES calculates all reactivity feedback terms (Doppler, sodium expansion, grid plate expansion, etc.). The bowing feedback term is calculated from a change in the average core outlet temperature using the correlation developed from static NUBOW calculations (actually, from CORTAC calculations which incorporates the NUBOW structural model).

E.2 Comment

Most of the reactivity trends shown in Figures E.2-2 and E.2-3 behave as expected. One that is more difficult to understand is the core axial expansion, particularly after 10 seconds when it goes positive. Please discuss this.

Response

In the unprotected loss of flow (ULOF) transient, for which the reactivity terms are shown in Figure E.2-3, the core axial feedback is initially negative but becomes positive after 10 seconds. This behavior is explained as follows. Initially the fuel heats up rapidly and then cools down as the power decreases while the cladding continues to heat up. Since the analysis conservatively assumes for this ULOF event that the fuel can shrink freely, the core axial expansion feedback based on the axial position of the fuel becomes positive. If the fuel were assumed to expand with the cladding (no slipping), the core axial expansion would be negative even after 10 seconds. Incidentally for the UTOP event in which the fuel expands more than the cladding, no slipping between the fuel and cladding is assumed for conservative calculations.

E.3 Comment

In the LOHS BDBE analysis covered in Section E.4, the primary pumps continue to function. It seems likely that a delayed trip of the primary pumps could aggravate the situation, as the natural circulation head would be reduced during the period after the IHTS is lost. Please explain the likely scenario once the IHTS is lost, i.e., at what time are signals to (1) scram the reactor, and (2) trip the primary pumps, likely? Have any transients in this class been examined?

Response

If the IHTS is lost (say at zero time), a scram signal is issued automatically at about 85 seconds when one of the trip parameters exceeds its setpoint. Then, the primary pumps are tripped after confirming that the flux is decreasing. Confirmation of this flux decrease will take 0.5 seconds at most.

If following the loss of IHTS cooling, the signals to the RPS fail to scram the reactor but trip the primary pumps, the resulting transient becomes essentially identical to the unprotected loss of flow event, which assumes loss of IHTS cooling as well as loss of primary flow. Peak temperatures for this event are all less than the conservative inherent safety criteria established.

E.4 Comment

The EBR-I reactor core was damaged by a bowing-related accident. Please explain the difference between the EBR-I, EBR-II, and PRISM reactors, and discuss the relevancy of the EBR-I accident to the current technology.

Response

See the ANL intra-laboratory memo of September 30, 1987, that follows.

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RESPONSES TO NRC COMMENTS
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INTRA-LABORATORY MEMO

September 30, 1987

To: Dean R. Pedersen RAS

From: Gerald H. Golden GJG EBR-II

Subject: Relevance of EBR-I Accident to Later LMRs

I. Background

In conjunction with the NRC review of the SAFR LMR reactor plant concept, the following request was made by the NRC staff:

"The EBR-I reactor was ultimately destroyed by a bowing-related accident. Please explain the difference between the EBR-I, EBR-II, and SAFR reactors, and make your case as to why the EBR-I accident is not relevant to the current technology."

A partial meltdown of the EBR-I Mark-II core took place in November 1955. After the cause of the problem was determined, the reactor and its fuel were appropriately modified and the Mark-III core was loaded in. EBR-I was then subjected to a rigorous series of tests lasting for more than a year. During this time there was no evidence of a positive component of feedback reactivity. Accounts are given in the literature of the EBR-I Mark-II partial meltdown[1,2,3] and of the subsequent stability tests on the Mark-III core.[1,3,4,5] Reference 3 is a definitive paper on the EBR-I experience.

It was well recognized that there were problems with the reactivity feedback in EBR-I, first with its Mark-I core and then with its Mark-II core.[1] Although the reactor was observed to operate smoothly at full power under steady-state conditions, power oscillations were instigated by small departures from normal operation. For example, right after a reduction in coolant flowrate, the power would increase, go through a maximum, and then decrease to a lower equilibrium value.[1,4] It was believed that there were two interacting feedback effects that produced this result. The first was a prompt-acting positive component that initially raised the power, and the second was a slower-acting but larger negative component that brought the power down to a new equilibrium value.[1,2,3]

The reactivity feedback characteristics of EBR-I began to receive considerable attention outside Argonne National Laboratory when the Mark-II core was installed in 1954, and the reactor was found to show a tendency toward spontaneous power oscillation under certain operating conditions. These feedback characteristics were reviewed in some detail at a general meeting on fast reactor safety in Detroit in December 1954. It was suggested at that meeting by H. Brooks and H. Hurwitz that the interacting feedback effects cited above might indeed be the cause of the power oscillations. They further suggested that oscillator experiments be undertaken on the reactor, in which its transfer function would be measured under various conditions of power and flow.[1]

II. EBR-I Partial Meltdown

In May 1955 an attempt was made to determine the feedback for the Mark-II core using rather crude equipment. The experiment was found not to be sufficiently accurate to allow a detailed investigation of the feedback to be made, although some useful information was obtained.[1]

Further oscillator testing with improved equipment was then undertaken in November 1955. It was decided to repeat an earlier test in which the reactor at zero coolant flow and low power (a few watts) was placed on about a 60-second period. The power was allowed to rise until the ratio of power P to time derivative of power $\dot{P}$ decreased to about 6 seconds, at which point the test was terminated.

The following were the conditions in the November 1955 tests that resulted in the partial meltdown:[2]

1. The operating pile period meter was disconnected from the scram circuit.

2. Two power level scram circuits were used, but with the trip set well above the normal operating power of 1150 kW, i.e., the level trips were in effect bypassed with regard to protection of the fuel from damage.

3. The reactor cooling system was in shutdown condition.

4. The reactor was made critical at about 11 watts.

5. The reactivity was increased via the control rods until the reactor was on about a 60 second period $(P/\dot{P})$. This was about 200 seconds into the transient, at which time the power was about 50 watts.

6. The power level increased at the 60 second period until it had reached 500 watts at 320 seconds into the transient.

7. At 497 seconds, the period was decreasing and the power level was approaching half of the normal operating value of 1150 kW.

8. What happened from this point on is quoted from a well-informed source:[6]

   "The understanding was that when the power increased to full level, the operator would scram the reactor. Important to note is the fact that all power and period-related trips were bypassed. The test engineers shouted to the operator,

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"scram." The operator panicked and pushed the wrong button. Instead of pushing the "big" scram button (which dropped $5 worth of reactivity), he pushed the junior scram button (which dropped only 10¢*). By that time the reactor was on a very short period of the order of one second, and the period was rapidly getting much shorter. Realizing the gravity of the matter, the test engineer physically shoved the operator aside and pushed the right scram button. But by that time it was too late. The power had increased to $10P_0$ and a substantial portion of the core had melted. Fortunately, damage was limited to the confines of the primary vessel, and only minor amounts of radioactivity (principally Kr and Xe) were released to the reactor building. Personnel were not affected in any way."

In retrospect, there were three interacting problems that led to the EBR-I partial meltdown:

1. the reactor design was inherently flawed;

2. the test was conducted much more casually than would be permitted (or considered) in the U.S. today; and

3. there was a major error by the reactor operator.

The remainder of this discussion is focused on the first of the above problems and on how the EBR-I Mark-II design differed from that of EBR-II and later LMRs.

III. Description of EBR-I

A brief description of EBR-I and its Mark-I and Mark-II core fuel elements is taken verbatim from Ref. 1:

"The reactor itself is shown in Fig. 1. The fuel elements are stainless steel tubes 0.448 in. OD with 0.020 in. wall thickness spaced on 0.494 in. centers. The tubes are positioned at the bottom by a tube plate, with triangular holes engaging a positioning pin at the bottom of the rod. Surrounding each triangular positioning hole in the tube plate are six coolant flow holes 3/16 in. in diameter and three 1/16 in. in diameter. The fuel elements are positioned above the core by a lower shield plate 4 in. thick, the elements passing through holes 0.460 in. in diameter. Approximately 4 in. above the lower shield plate the fuel elements pass through a similarly drilled seal plate, 2 in. thick, with additional shield plates, either 2 or 4 in. thick spaced 4 in. apart, above the seal plate for approximately 75 additional inches to the top plate, from which the fuel elements are supported. A clearance of approximately 1/4 in. exists between the bottom of the fuel element and the tube plate.

*This should be about 39¢.

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"The fuel bearing section of the element is the lower 21 in.; above this is the "handle" which is solid stainless steel, fluted at the lower shield and seal plates to permit flow.

"In the Mark-I loading the fuel bearing section of each element had two spacer ribs 0.042 in. high located so that when the triangular positioning tips of the elements were engaged in the matching holes in the tube plate, the ribs were brought in line with neighboring rods. Actually a 0.004 in. nominal clearance was allowed, as can be seen from the above dimensions.

"In the Mark-II loading the spacer ribs were omitted and the positioning tip was made cylindrical, thus permitting 0.046 in. separation of the rods.

"Highly enriched uranium in each fuel element was in the form of cylinderical slugs. In the Mark-I core four slugs 0.364 in. in diameter and 1-7/8 in. in length were loaded in most rods. Some were loaded with slugs 0.384 in. in diameter by 2.5 in. long. Below the fuel, a 4-1/2 in. long natural uranium slug was loaded as part of the lower blanket, and above the fuel an 8 in. natural uranium slug was loaded as the top blanket. In the Mark-II loading the fuel slugs were U-2% Zr alloy, 4-1/4 in. long by 0.384 in. in diameter, with a lower blanket slug 4-1/4 in. long and an upper blanket slug 8 in. long. The various slugs were separated and positioned in the tubes by 0.005 in. stainless steel spacers. The annulus between slug and fuel tube was filled with NaK as a heat transfer bond.

"The inner blanket consists of stainless steel tubes of 0.020 in. wall thickness drawn on 15/16-in. diameter natural uranium rods, 20-1/4 in. long. These rods too are supported at the top plate, and hang from a handle which passes through the various shield plates, the uranium blanket section being positioned by the lower shield plate and the lower tube plate.

"The entire core-inner blanket assembly is contained in a double-walled reactor tank which at the core level is 15.87 in. ID by 28 in. long. This inner tank is surrounded by an outer, air-cooled blanket which consists of 84 stainless steel jacketed, keystone-shaped natural uranium-bricks stacked seven high to form a 'cup.' The cup is supported on a hydraulic lift and may be raised to surround the core-inner blanket assembly or dropped below the core to stop the chain reaction.* Mounted on the hydraulic lift and operating through cylindrical holes in the blankets are twelve 2-in. natural uranium, stainless steel jacketed

* This gives the "big" scram, worth about $5.

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rods, 8 designated as safety rods** and 4 designated as control rods. Below the reactor tank is a uranium safety block which may be dropped out to provide additional shutdown.**

"Figure 2 shows a plan view of the reactor at the core midplane, showing the reactor core, hexagonal separator, inner blanket, reactor tank, and cup. The same inner blanket and cup were used for the Mark-I and Mark-II loadings.

"Coolant flow through both Mark-I and Mark-II cores was in series through inner blanket and core. Referring to Fig. 1, coolant enters the plenum between the lower shield plate and the next higher plate (the seal plate), flows down through the inner blanket to a plenum below the lower tube plate, then up through the core past the lower shield plate and seal plate and out above the seal plate." See Fig. 3.

IV. Cause of EBR-I Instability

Before discussing the cause of the EBR-I instability, it is important to emphasize the following features of this reactor with its Mark-I and Mark-II cores:

1. The core was very small, being about 7.5 in. across flats by about 8.5 in. high. The average core power density was about 190 kW/liter.

2. The radial flux (hence power) distribution in the core fell off rapidly with increasing distance from the core centerline.

3. On the Mark-I fuel element there was minimal provision for maintaining spacing between elements, and on the Mark-II design there was no such provision. In the Mark-II design there was 0.046 in. separation between the fuel elements. Moreover, neither type of element was contained in a subassembly. See Fig. 4.

4. For both the Mark-I and Mark-II designs, individual elements were supported at the bottom by the tube plate and at the top by the lower shield plate, which was 4 in. thick. See Figs. 5 and 6.

As noted in Section I, Background, it was early recognized that there were two competing reactivity effects in the Mark-I and Mark-II cores that produced the observed power instabilities. The first effect was fast-acting and positive, and the second effect was slower, negative, and larger in absolute magnitude. The causes of these two effects were conclusively demonstrated and reported in Ref. 3. Briefly, the fast positive effect was due to bowing of fuel elements radially inward and the slower and larger negative effect was due to time-dependent

** These give the faster-acting (100 millisecond) junior scram, worth about 39¢.[7]

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thermal expansion of the thick lower shield plate. As noted earlier, under the test conditions leading to the partial meltdown, there was no reactor coolant flow. Thus, as the power increased, the inboard sides of the elements heated and expanded more rapidly than the outboard sides, due to the steep radial power distribution in the core. The selective expansion led to the rapid radially-inward bowing once the fuel element temperatures began to increase. This effect led to the partial meltdown.

On somewhat longer time scales, the thermal expansion of the lower shield plate led to increased separation between fuel elements, a negative reactivity effect. This situation is illustrated in Figs. 7 and 8. Figure 7 shows the fuel element orientation under isothermal conditions, and Fig. 8 shows the orientation during a power rise at zero flowrate. Note that the initial downward bowing of the 4 in. thick plate as it is heated at its bottom tends to further push out the elements.

V. Redesign of EBR-I and Its Mark-III Fuel Element

To correct the above problems, both the fuel element and the above-core structure of EBR-I were modified.[1] The reactor itself was modified to eliminate the lower shield plate, as shown in Fig. 9. The Mark-III fuel element was modified mainly by going to a zirconium alloy cladding having three symmetrically located spacer wires on it, and then placing 36 such elements in a hexagonal can. The resulting core arrangement is shown in Fig. 10.

VI. Stability of the Mark-III Core

Extensive stability testing of EBR-I with its Mark-III core was initiated in November 1957 and continued for more than a year. During this time there was no evidence of a positive component of the feedback reactivity or of power instability.[1,3,4,5] Quoting directly from Ref. 4:

> "In summary, those features responsible for the instability of Mark-II, namely, the prompt positive and delayed negative power coefficient components, were unquestionably the result of design peculiarities. The elimination of the perforated shield plate system coupled with the addition of stabilizing ribs to the Mark-III fuel rods has resulted in a reactor whose performance is unquestionably stable under all credible operation conditions."

VII. Conclusions

The conclusions that follow from the foregoing sections are:

1. The mechanisms leading to the partial meltdown of the EBR-I Mark-II core are well understood.

2. All fast reactors subsequent to EBR-I have been designed to preclude the fast-acting positive reactivity feedback effect observed in that reactor.

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3. No evidence of significant fast-acting positive reactivity feedback has been observed in any operating fast reactor subsequent to EBR-I with its Mark-II core.

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1. F. W. Thalgott, et al., "Stability Studies on EBR-I," Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1 September - 13 September 1958, 12, Reactor Physics, pp. 242-266.

2. R. O. Brittan, "Analysis of the EBR-I Core Meltdown," Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1 September - 13 September 1958, 12, Reactor Physics, pp. 267-272.

3. R. R. Smith et al., "An Analysis of the Stability of EBR-I, Marks I to III, and Conclusions Pertinent to the Design of Fast Reactors," Proceedings of the Seminar on the Physics of Fast and Intermediate Reactors, Sponsored by the International Atomic Energy Agency and Held in Vienna, 3-11 August 1961, III, SM-18/49, pp. 43-83, IAEA, Vienna 1962.

4. R. R. Smith, et al., "Instability Studies with EBR-I, Mark III," ANL-6266, December 1960.

5. L. J. Koch, W. B. Loewenstein, and H. O. Monson," Addendum to Hazard Summary Report, Experimental Breeder Reactor-II (EBR-II), Appendix D, Stability Studies on EBR-I, Mark-III Core, June 1962,

6. R. R. Smith, private communication.

7. H. V. Lichtenberger et al., "Operating Experience and Experimental Results Obtained from an NaK-Cooled Fast Reactor," Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Geneva, 8 August-20 August 1955, 3, pp. 345-360.

GHG:jm

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RESPONSES TO NRC COMMENTS
ON PSID APPENDIX E

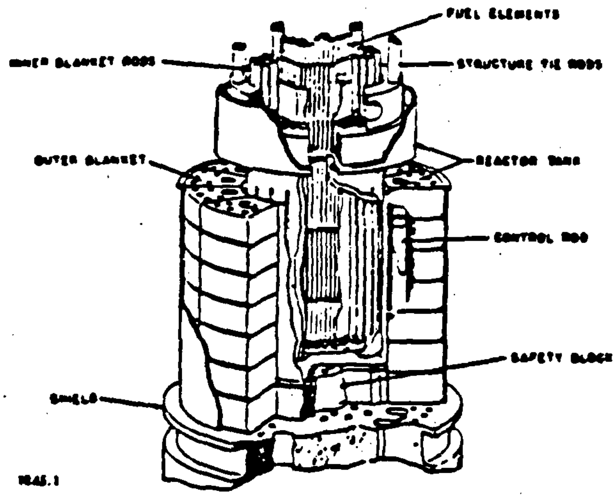


Figure 1. Cutaway view of EBR-I, Mark I and Mark II loadings

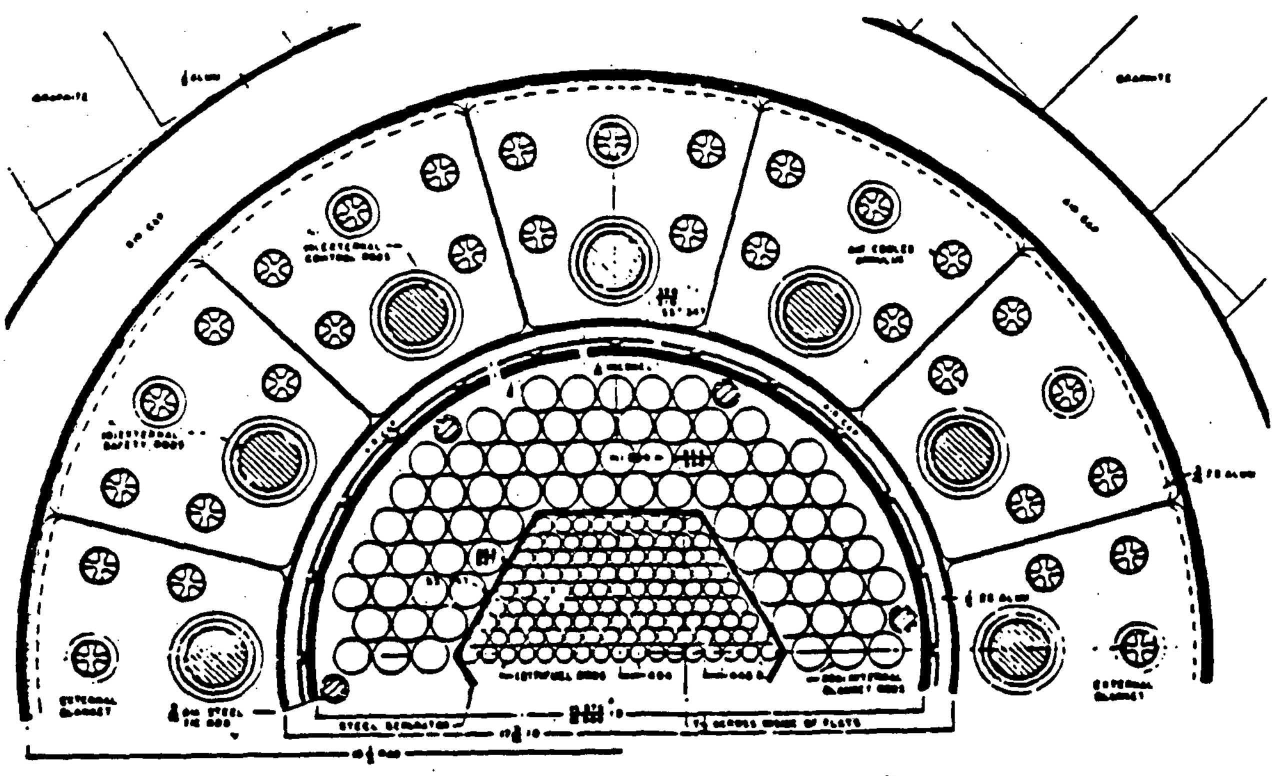


Figure 2. Plan view of the core, Mark I and Mark II

Above figures taken from reference 1.

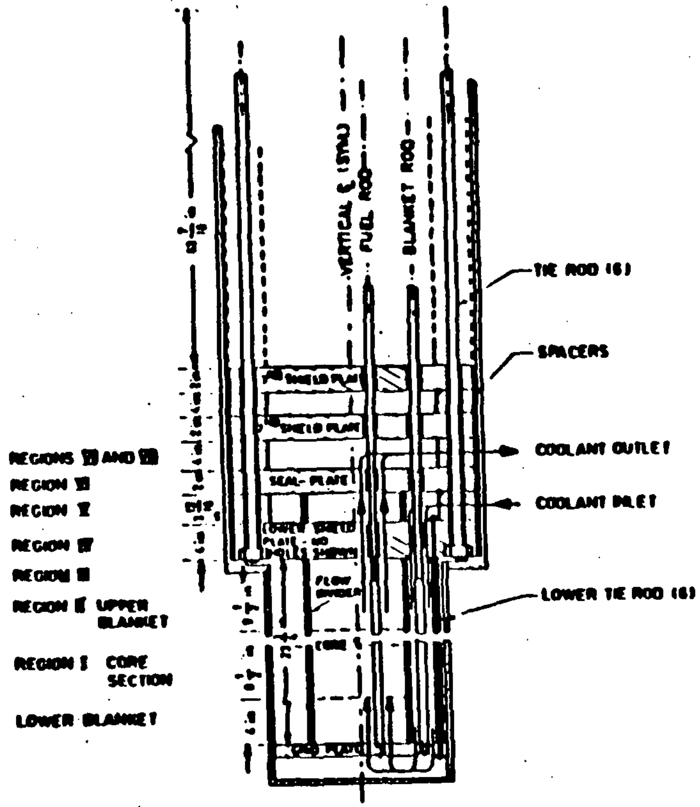


Fig. 3
Schematic view of EBR-I structure.

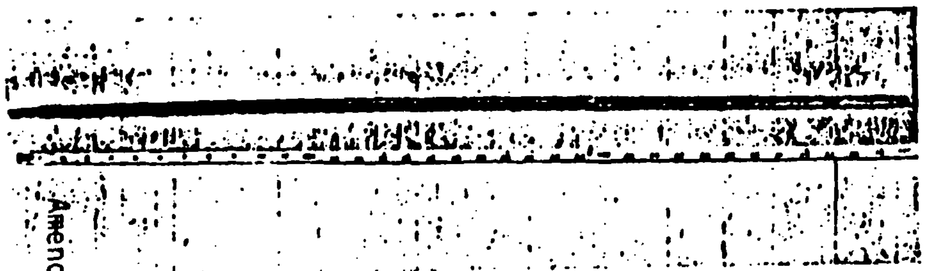


Fig. 4
EBR-I, Mark II fuel rod.

Above figures taken from reference 3.

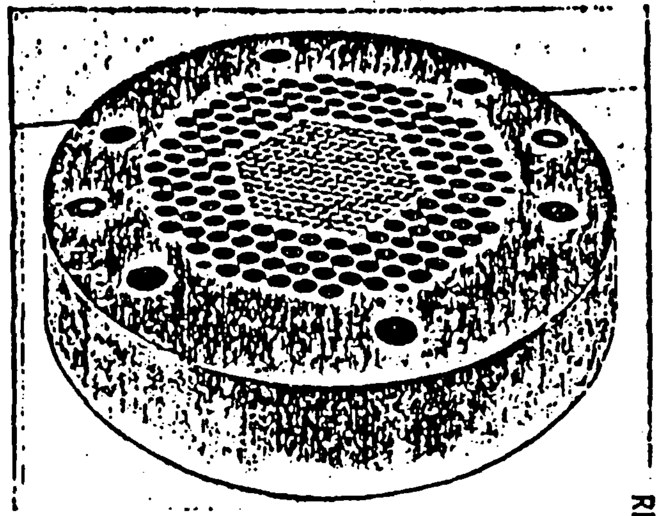


Fig. 5
EBR-I shield plate.

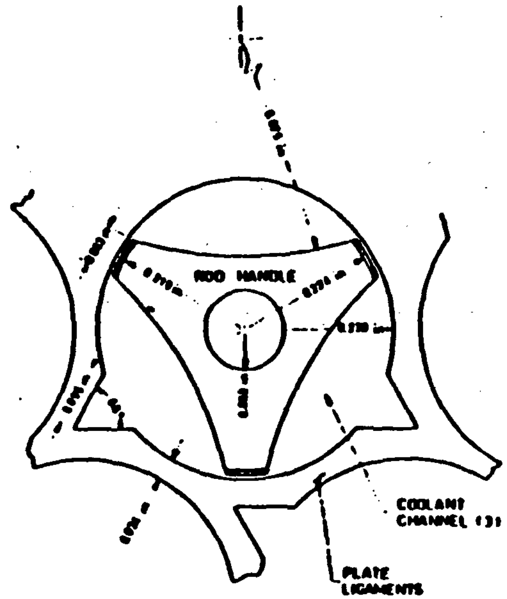


Fig. 6
Horizontal cross-section through lower shield plate and rod handle.

RESPONSES TO NRC COMMENTS
ON PSID APPENDIX E

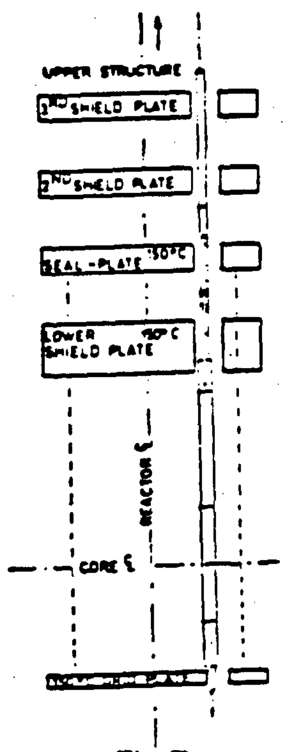


Fig. 7
Typical fuel-rod orientation at 150°C, zero power. Rods are preferentially gathered at the inner edge of grid and lower shield plates because of radial expansion.

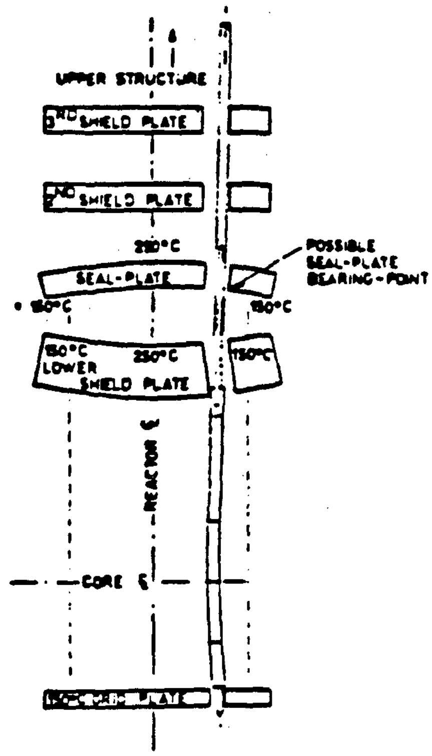


Fig. 8
Typical fuel-rod orientation during increasing portion of oscillating power cycle or immediately following a step-power increase.

Above figures taken from reference 3.

RESPONSES TO NRC COMMENTS
ON PSID APPENDIX E

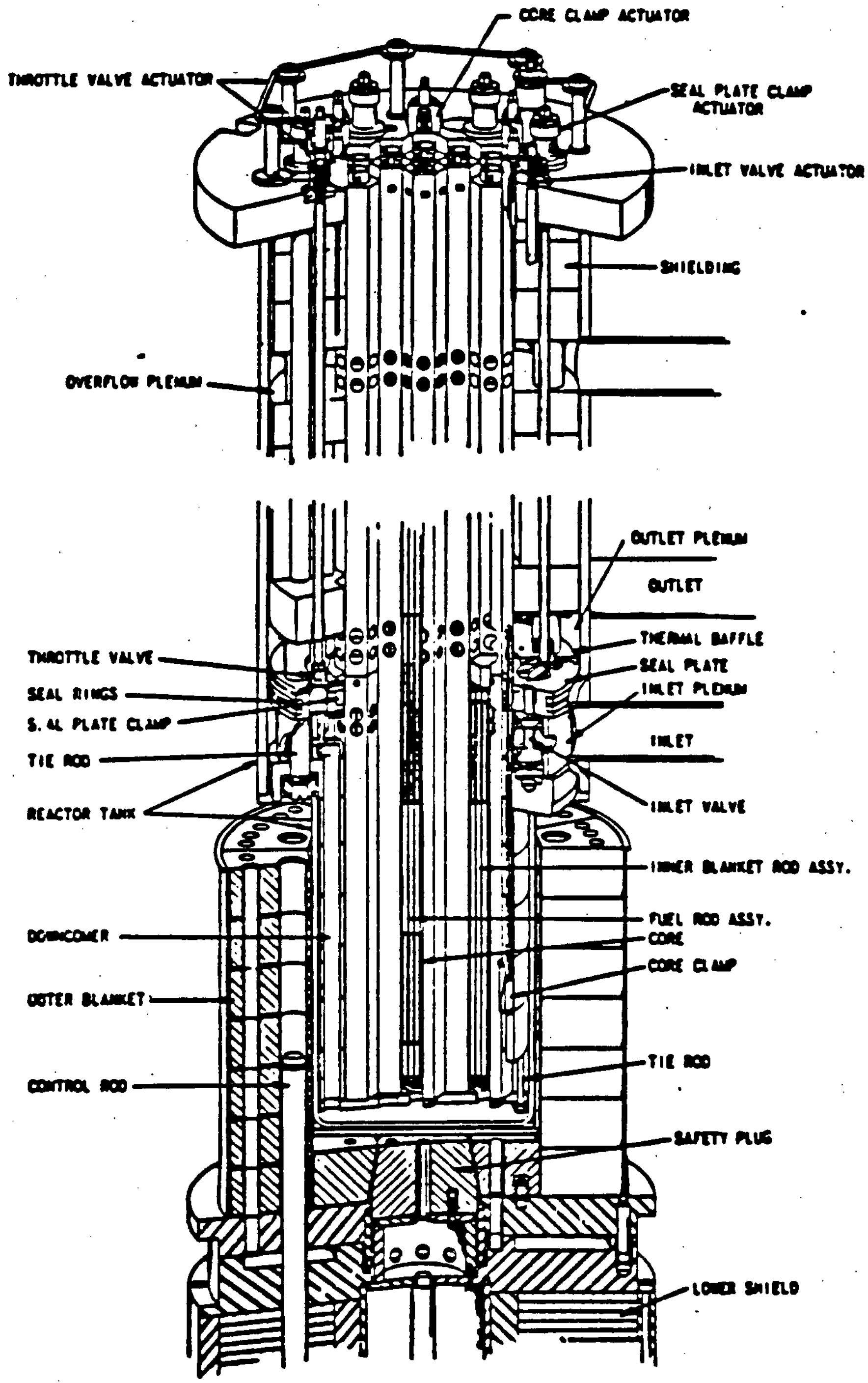


Figure 9 EBR-I, Mark III inner tank assembly

Above figure taken from reference 1.

RESPONSES TO NRC COMMENTS
ON PSID APPENDIX E

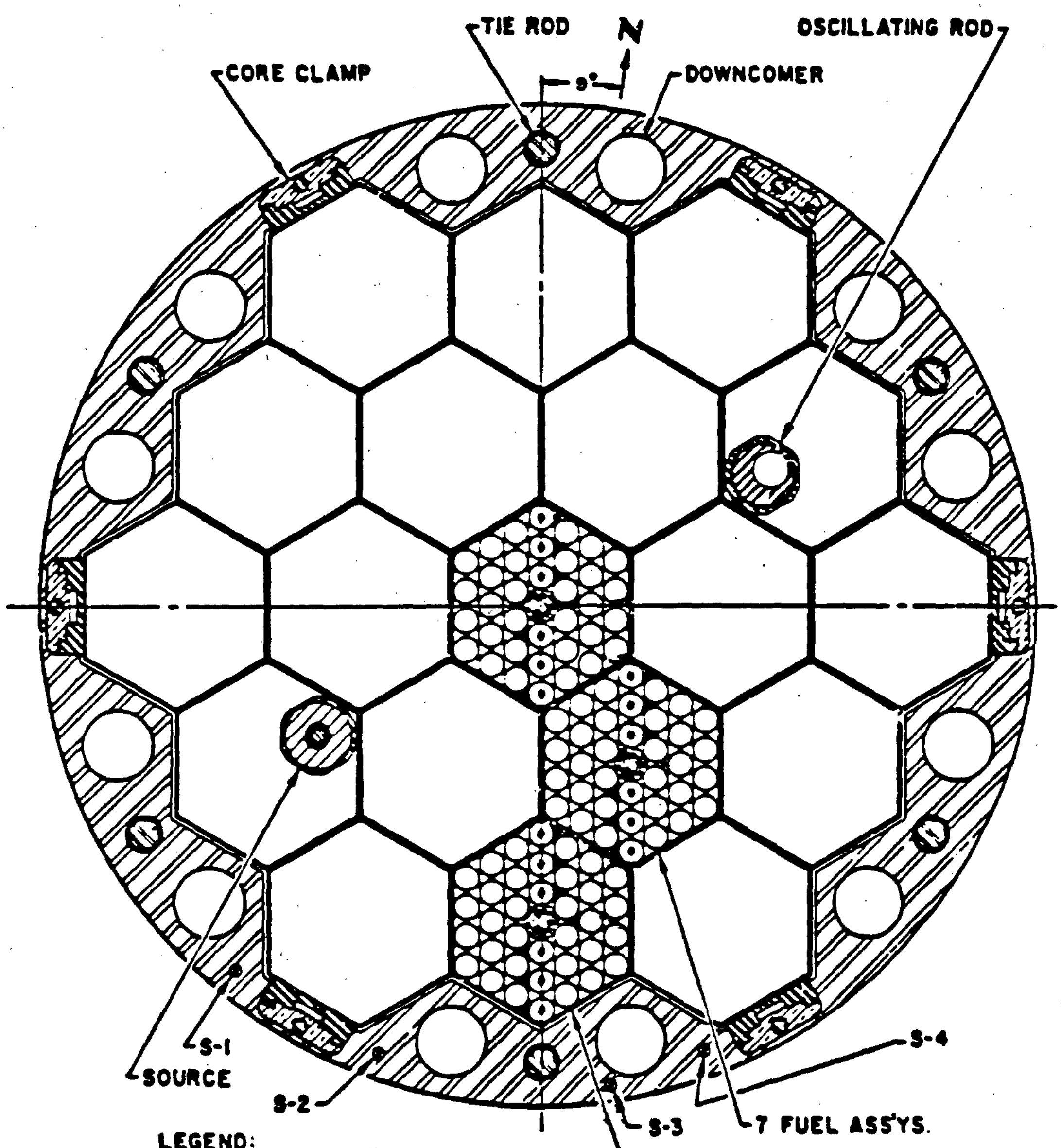


Figure 10. Plan view of EBR-I, Mark III at core centerline

Above figure taken from reference 1.

RESPONSES TO NRC COMMENTS ON PSID APPENDIX E

E.5 Comment

It appears that a cable breakage, instantaneously cutting power to one of the EM pumps, is the worst case of LOF BDBE, in that the reduction in flow to the reactor core is very quick. Can PRISM survive such an event (with and without scram) without heating the sodium to the boiling temperature?

Response

The event in question, the unprotected (without scram) loss of one primary pump with the IHTS operating, has recently been analyzed. Peak temperatures for this event are all less than the conservative inherent safety criteria established, as summarized below.

	Nom. Peak °F	Limit °F
Average core outlet	960	1300 (Design)
Peak assembly outlet	1130	1600
Peak cladding midwall	1140	1450
Peak fuel-clad interface (long-term)	1050	1290
Peak fuel temperature	1390	1720

E.6 Comment

Please provide detailed analysis plots for the first 50-60 seconds for all results reported in the analyses in the PSID.

Response

Detailed plots for the beyond-design-basis events noted below are attached: (all events are without scram)

Loss of Flow and Loss of IHTS Cooling	Figs. E.6-1a through E.6-1e
Loss of IHTS Cooling	Figs. E.6-2a through E.6-2e
Single Rod Withdrawal	Figs. E.6-3a through E.6-3e
Loss of One Primary Pump	Figs. E.6-4a through E.6-4e
All (Six) Rods Withdrawal	Figs. E.6-5a through E.6-5e

E.7 Comment

Provide a schedule for submitting the existing EBR-II data to support the assumed Doppler reactivity effect.

Response

ANL is using the EBR-II Shutdown Heat Removal tests to validate the calculational sequences (neutronics through plant dynamics) used in the design and analysis of advanced LMR's and expects the data to be available in the Spring of 1989.

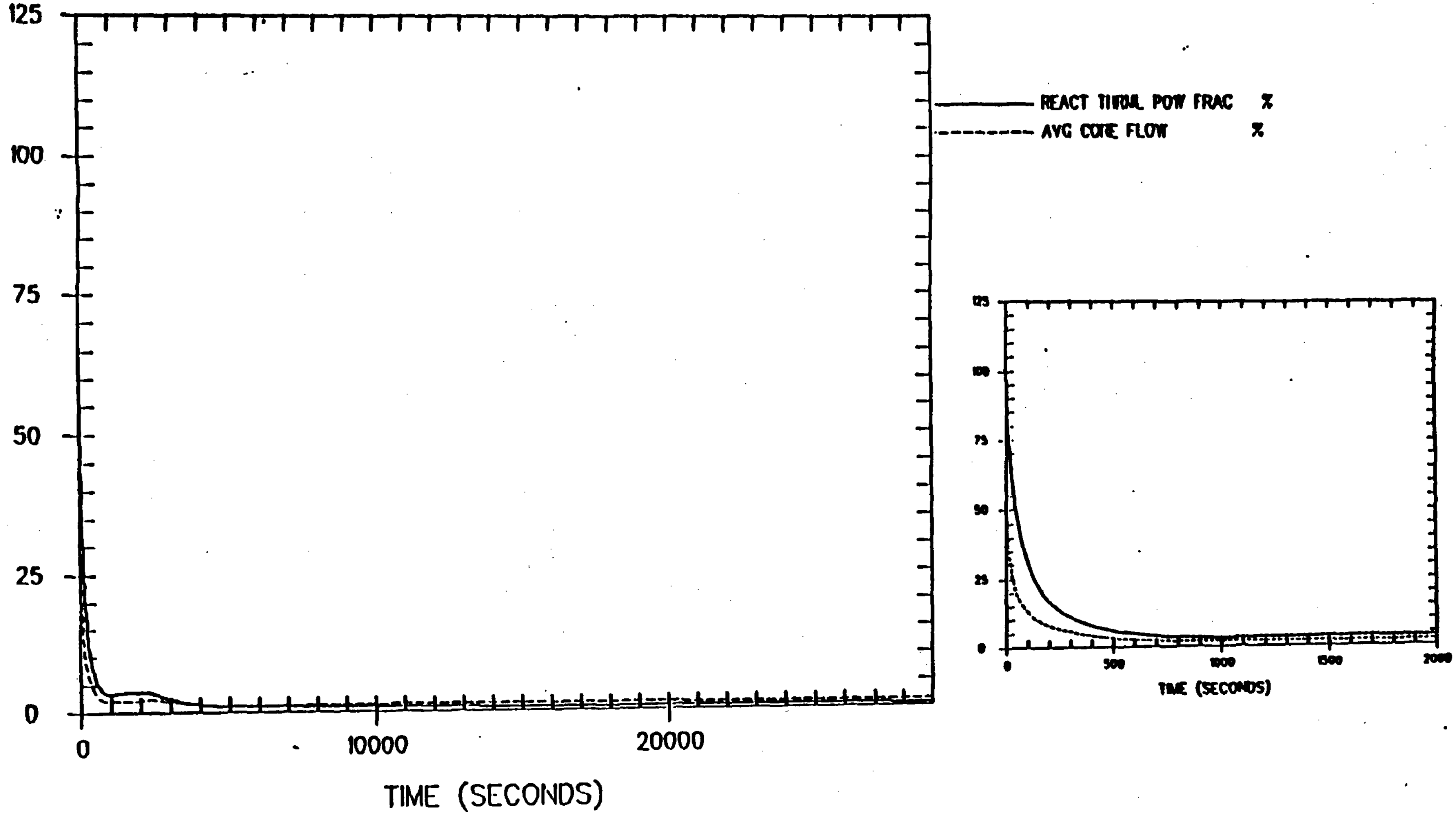


FIGURE E.6-1a REACTOR POWER AND PRIMARY FLOW FOLLOWING TRIP OF ALL PRIMARY PUMPS AND LOSS OF IHTS, COMBINED WITH FAILURE TO SCRAM (ULOF)

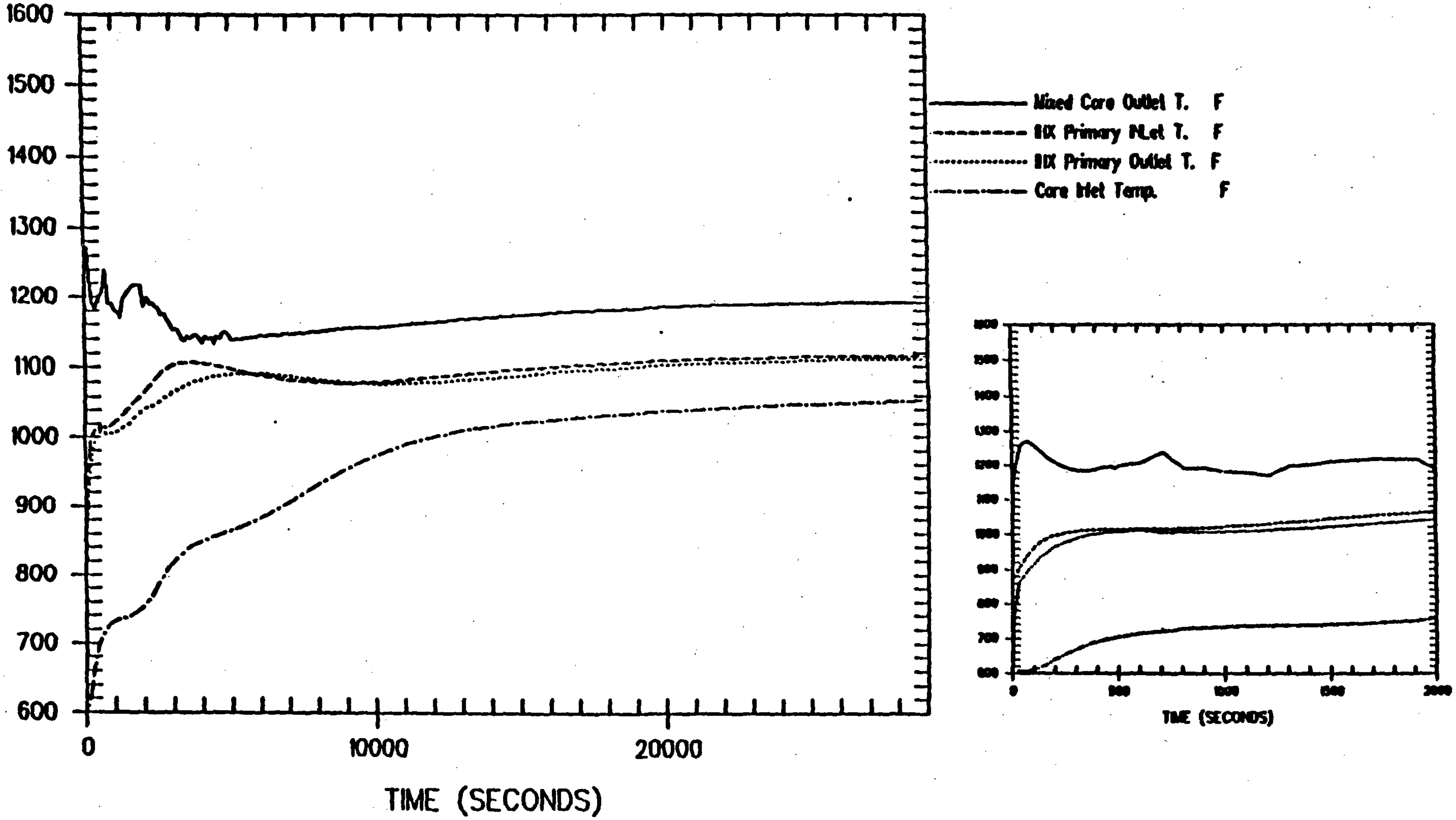


FIGURE E.6-1b PRIMARY SODIUM TEMPERATURES FOLLOWING TRIP OF ALL PRIMARY PUMPS AND LOSS OF IHTS, COMBINED WITH FAILURE TO SCRAM (ULOF)

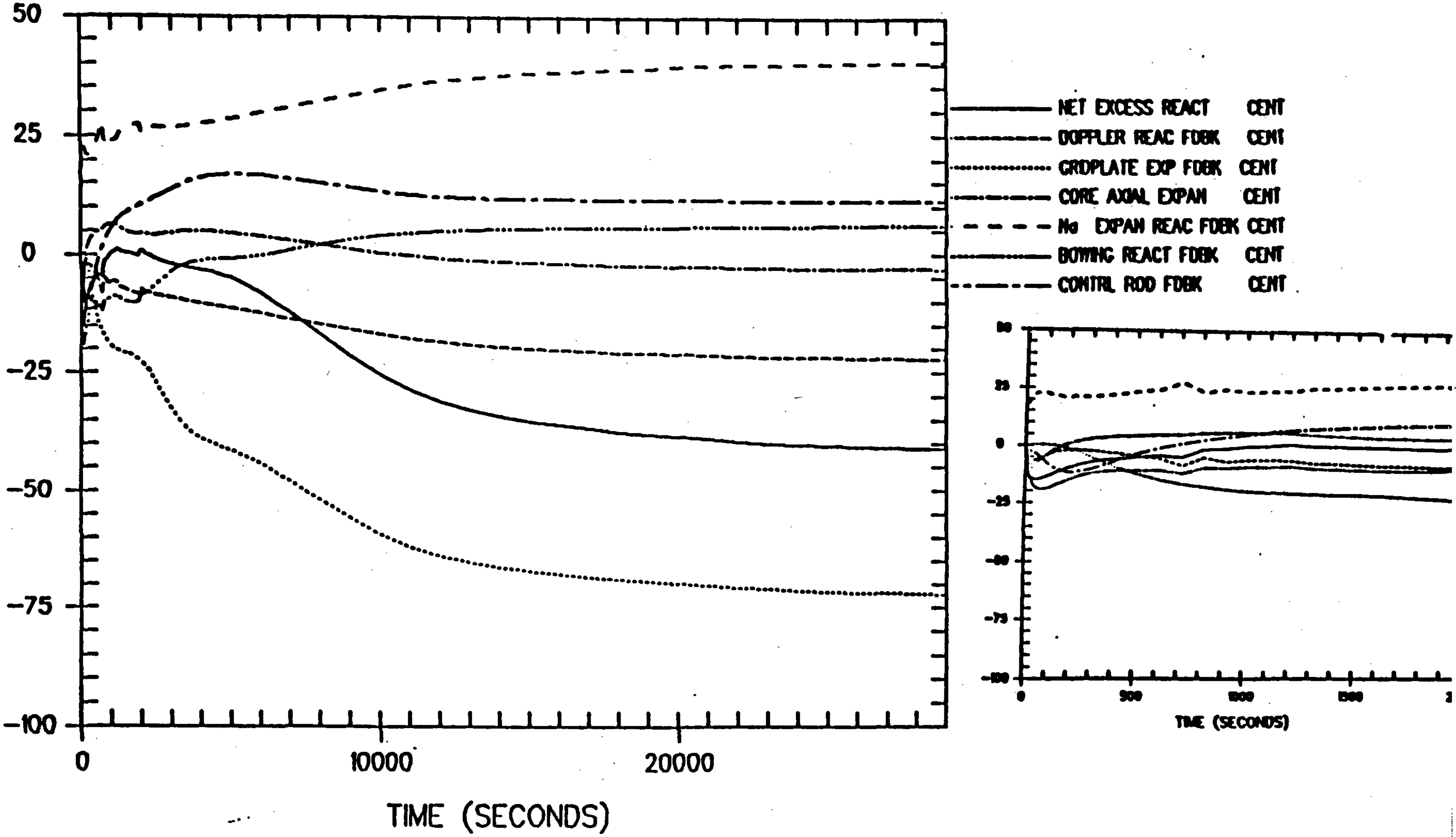


FIGURE E.6-1c CORE INHERENT REACTIVITY FEEDBACK FOLLOWING TRIP OF ALL PRIMARY PUMPS AND LOSS OF IHTS, COMBINED WITH FAILURE TO SCRAM (ULOF)

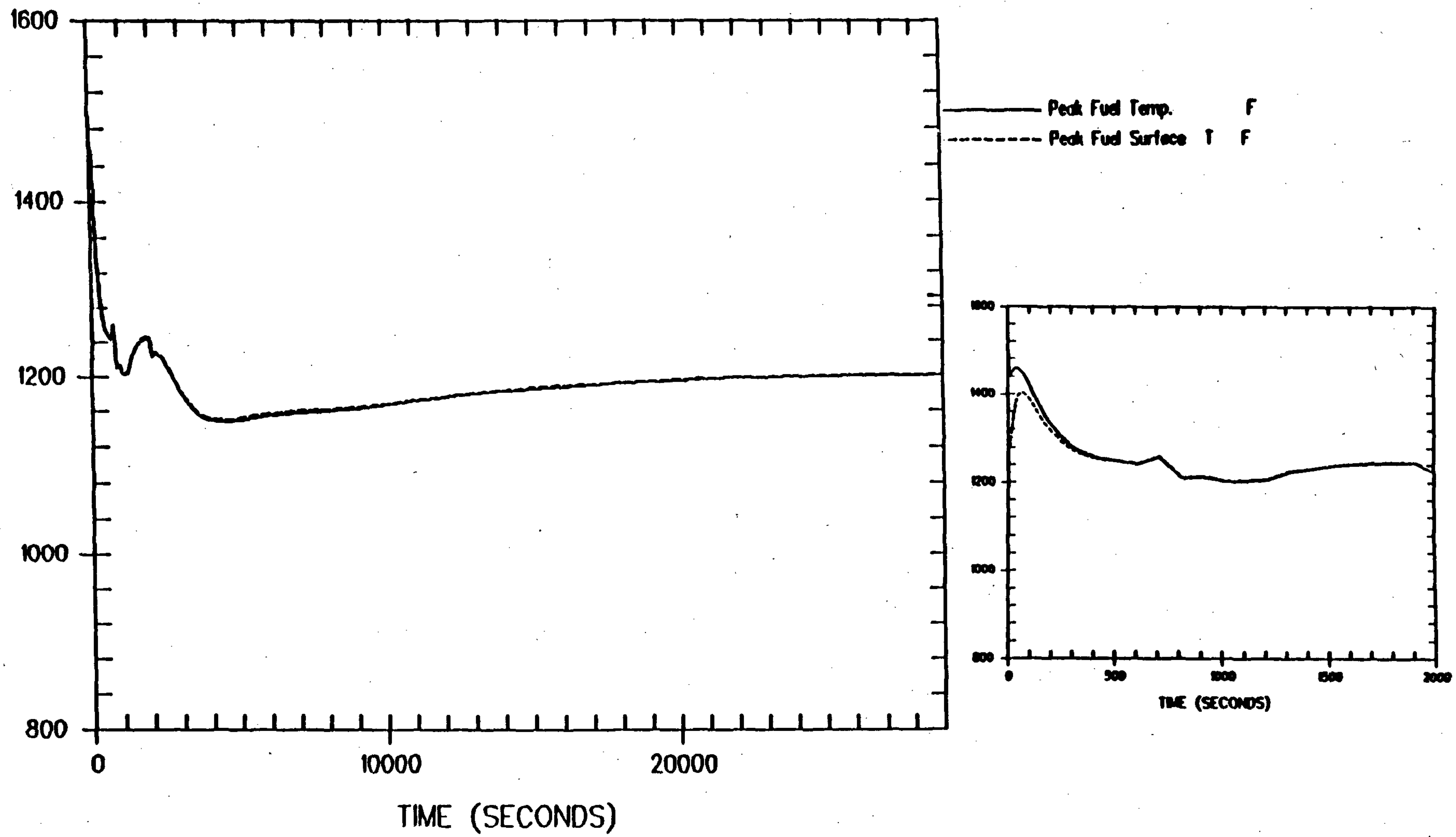


FIGURE E.6-1d PEAK FUEL CENTERLINE AND SURFACE TEMPERATURES FOLLOWING TRIP OF ALL PRIMARY PUMPS AND LOSS OF IHTS, COMBINED WITH FAILURE TO SCRAM (ULOF)

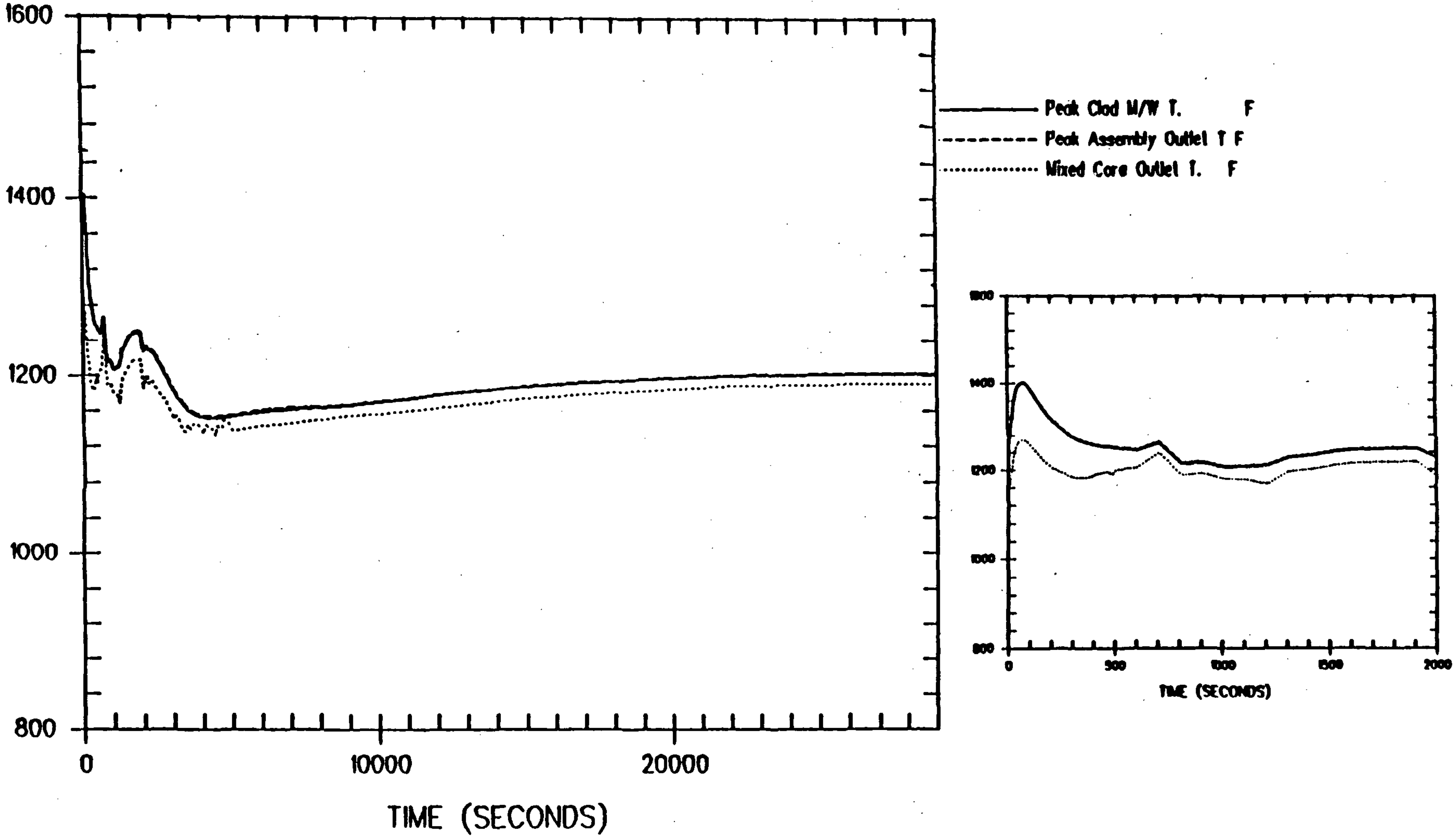


FIGURE E.6-1e PEAK CLADDING AND SODIUM TEMPERATURES AND MIXED CORE OUTLET TEMPERATURE FOLLOWING TRIP OF ALL PRIMARY PUMPS AND LOSS OF IHTS, COMBINED WITH FAILURE TO SCRAM (ULOF)

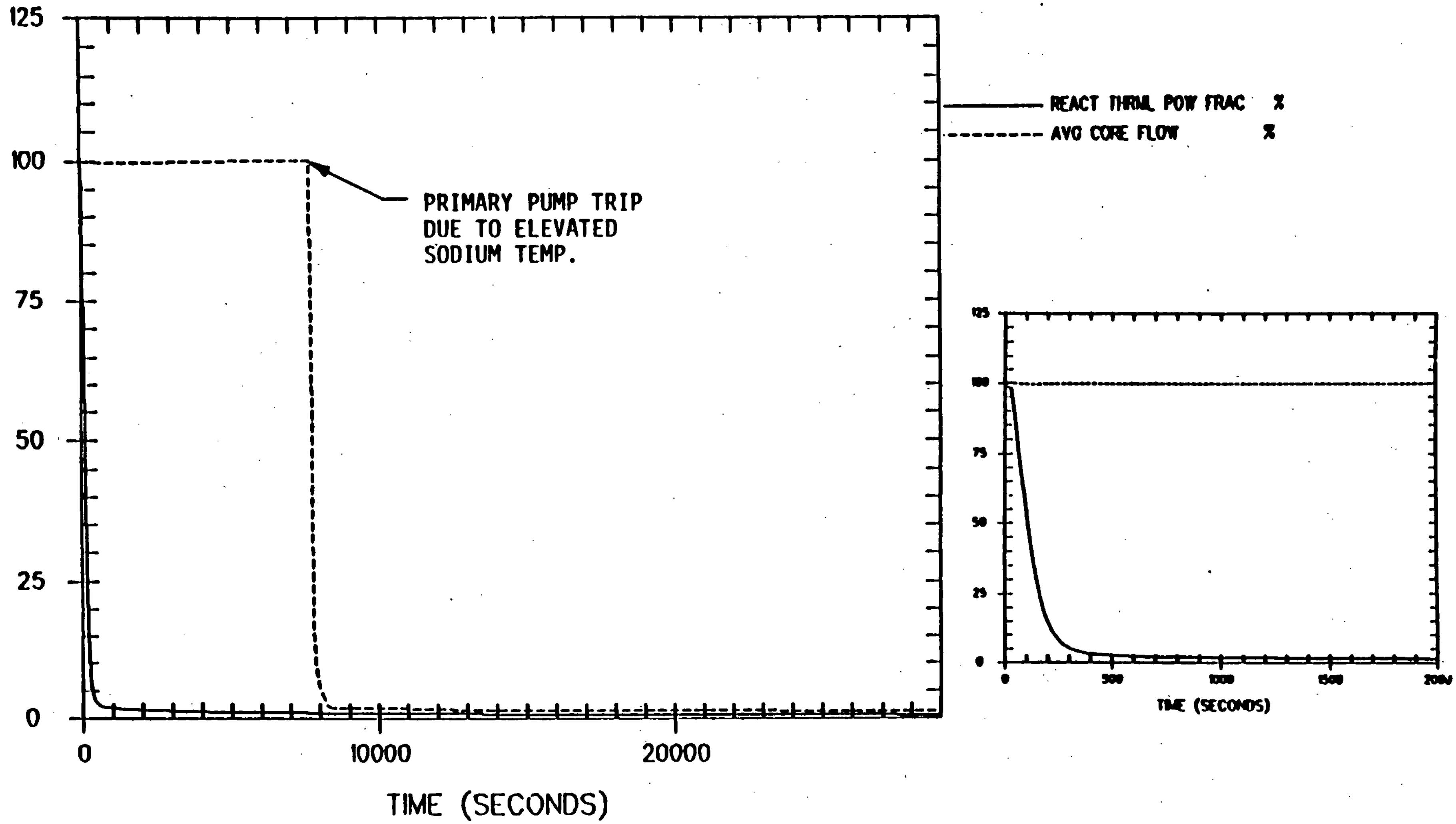


FIGURE E.6-2a REACTOR POWER AND PRIMARY FLOW FOLLOWING LOSS OF IHTS COOLING WITH FAILURE TO SCRAM (ULOHS)

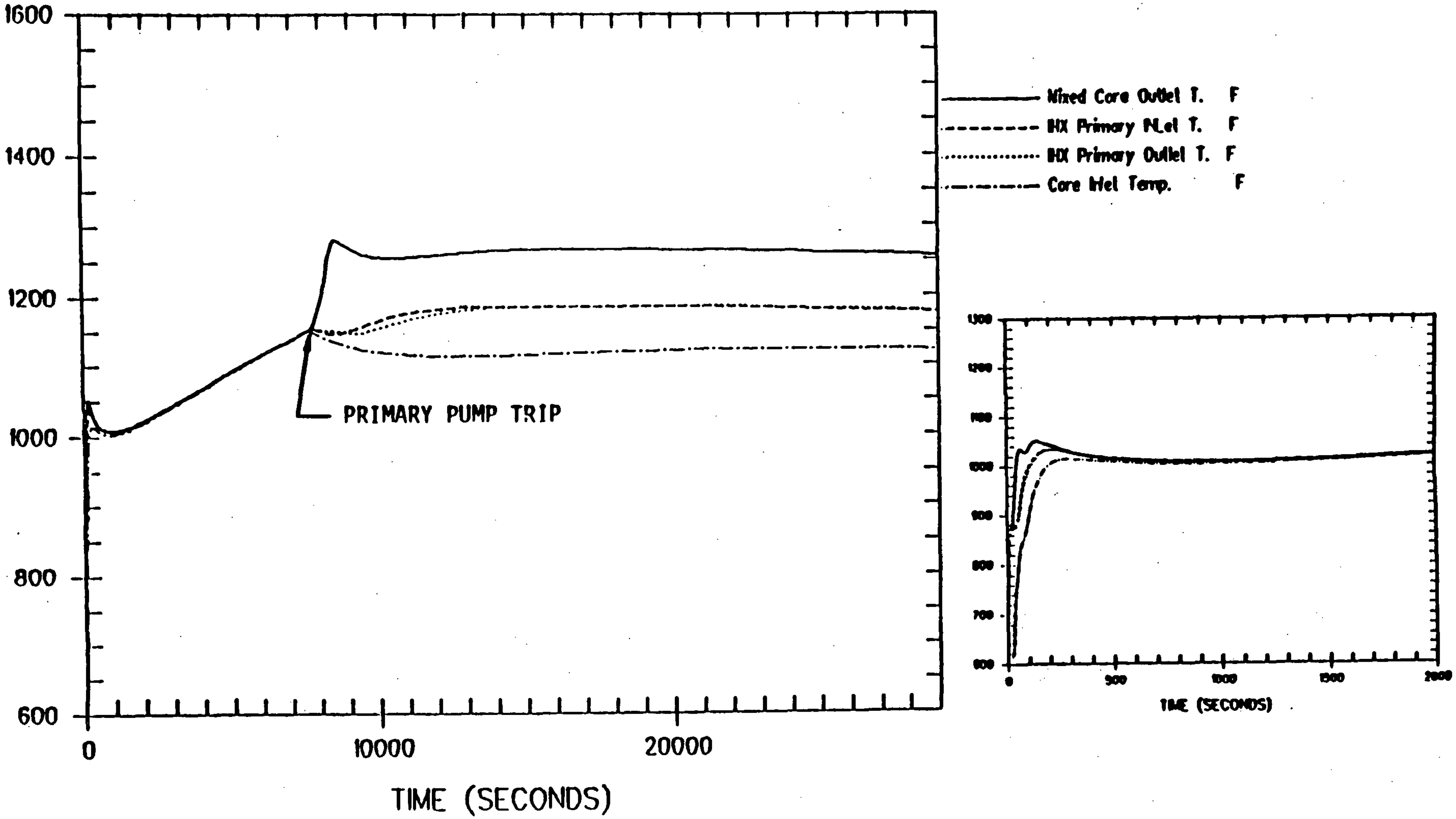


FIGURE E.6-2b PRIMARY SODIUM TEMPERATURES FOLLOWING LOSS OF IHTS COOLING WITH FAILURE TO SCRAM (ULOHS)

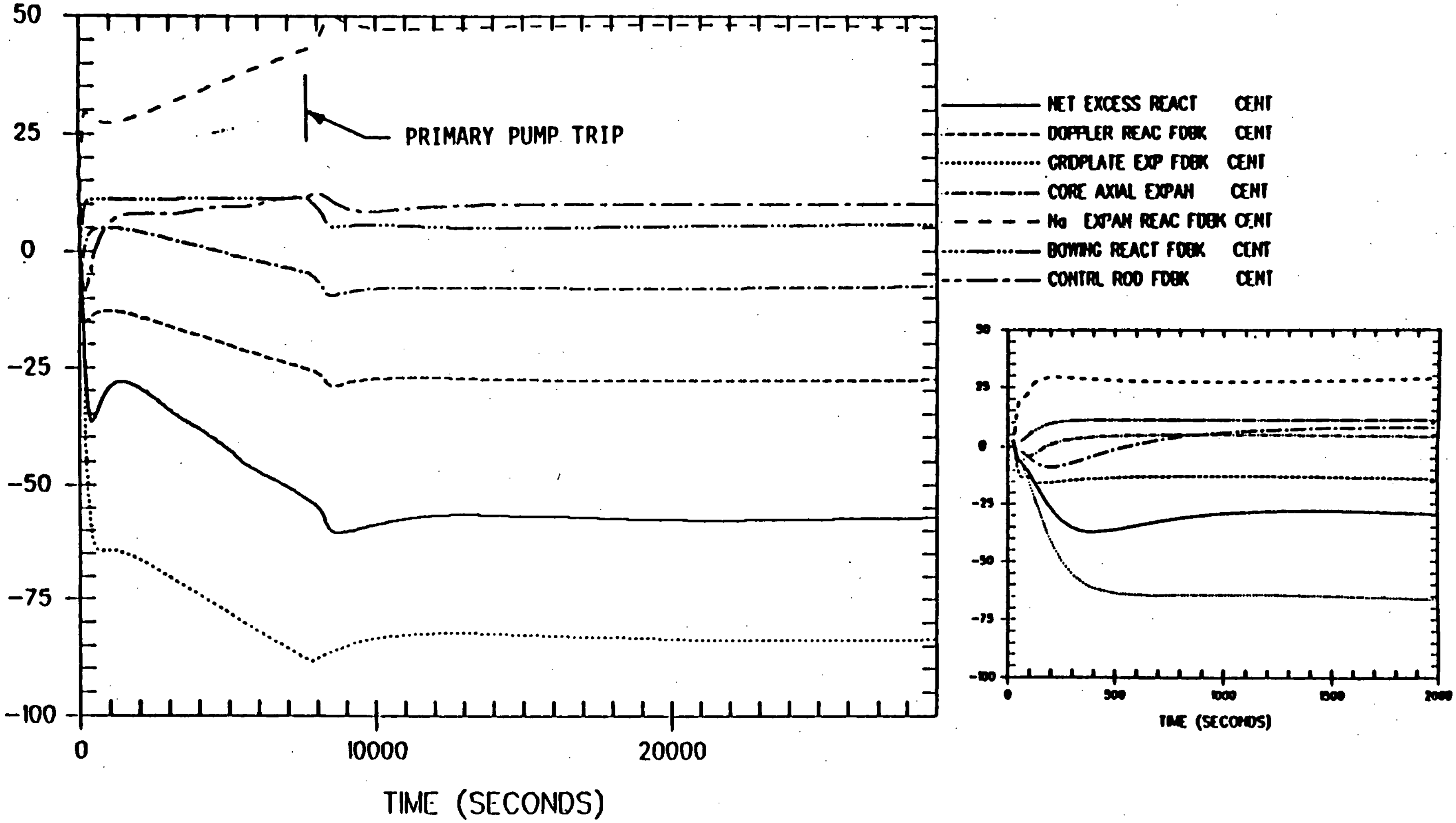


FIGURE E.6-2c CORE INHERENT REACTIVITY FEEDBACK FOLLOWING LOSS OF IHTS COOLING WITH FAILURE TO SCRAM (ULOHS)

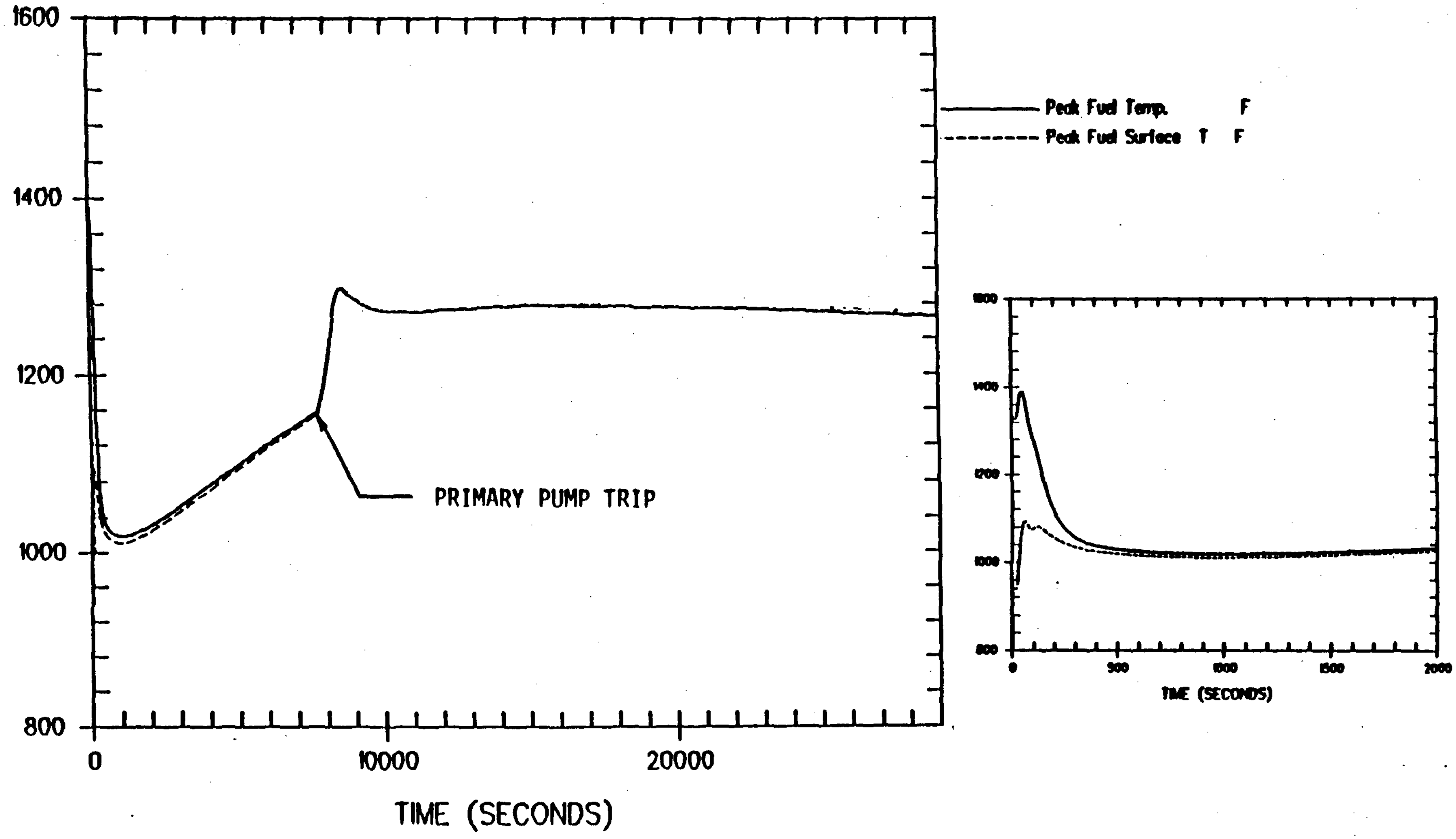


FIGURE E.6-2d PEAK FUEL CENTERLINE AND SURFACE TEMPERATURES FOLLOWING LOSS OF IHTS COOLING WITH FAILURE TO SCRAM (ULOHS)

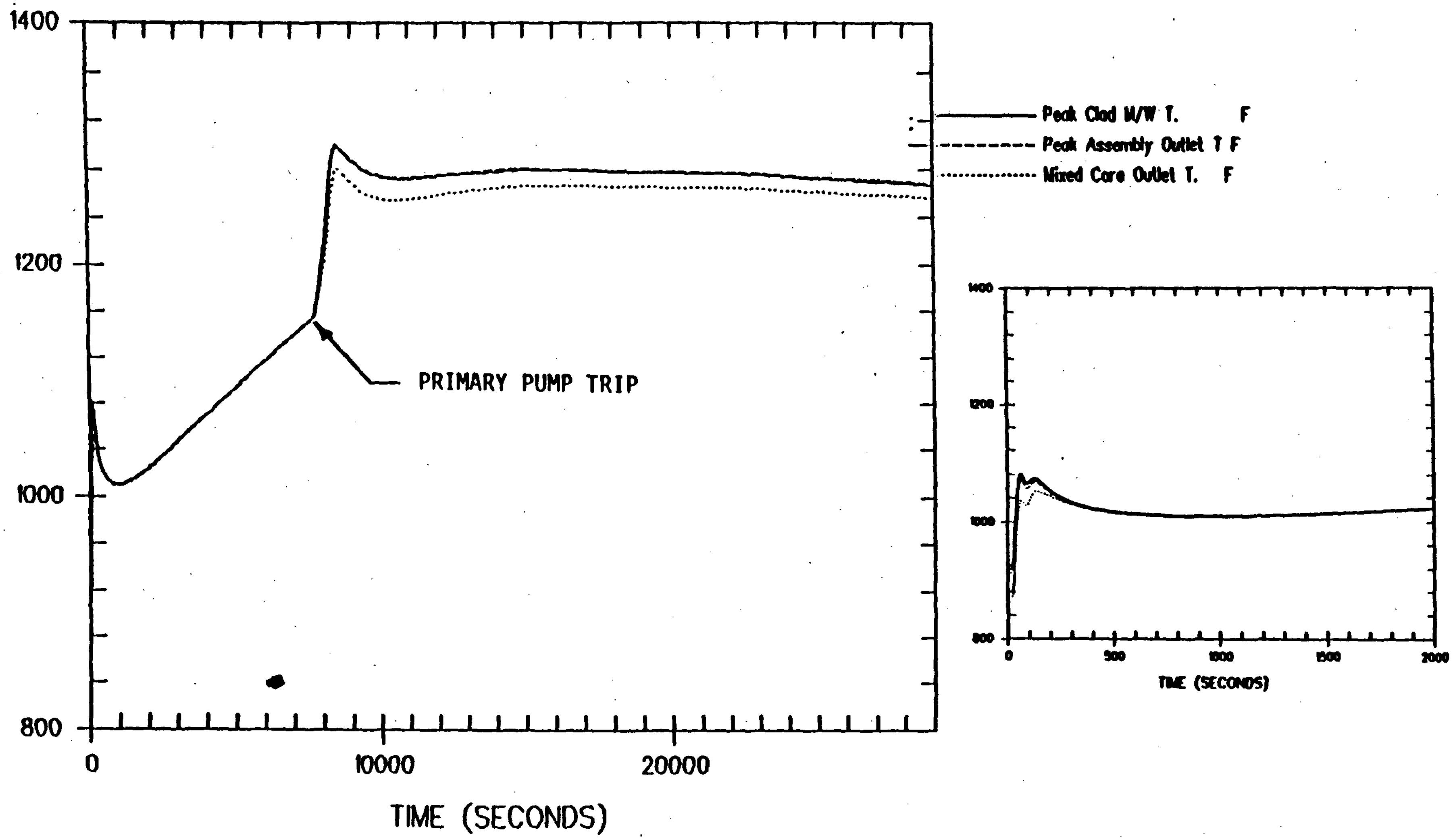


FIGURE E.6-2e PEAK CLADDING AND SODIUM TEMPERATURES AND MIXED CORE OUTLET TEMPERATURE FOLLOWING LOSS OF IHTS COOLING WITH FAILURE TO SCRAM (ULOHS)

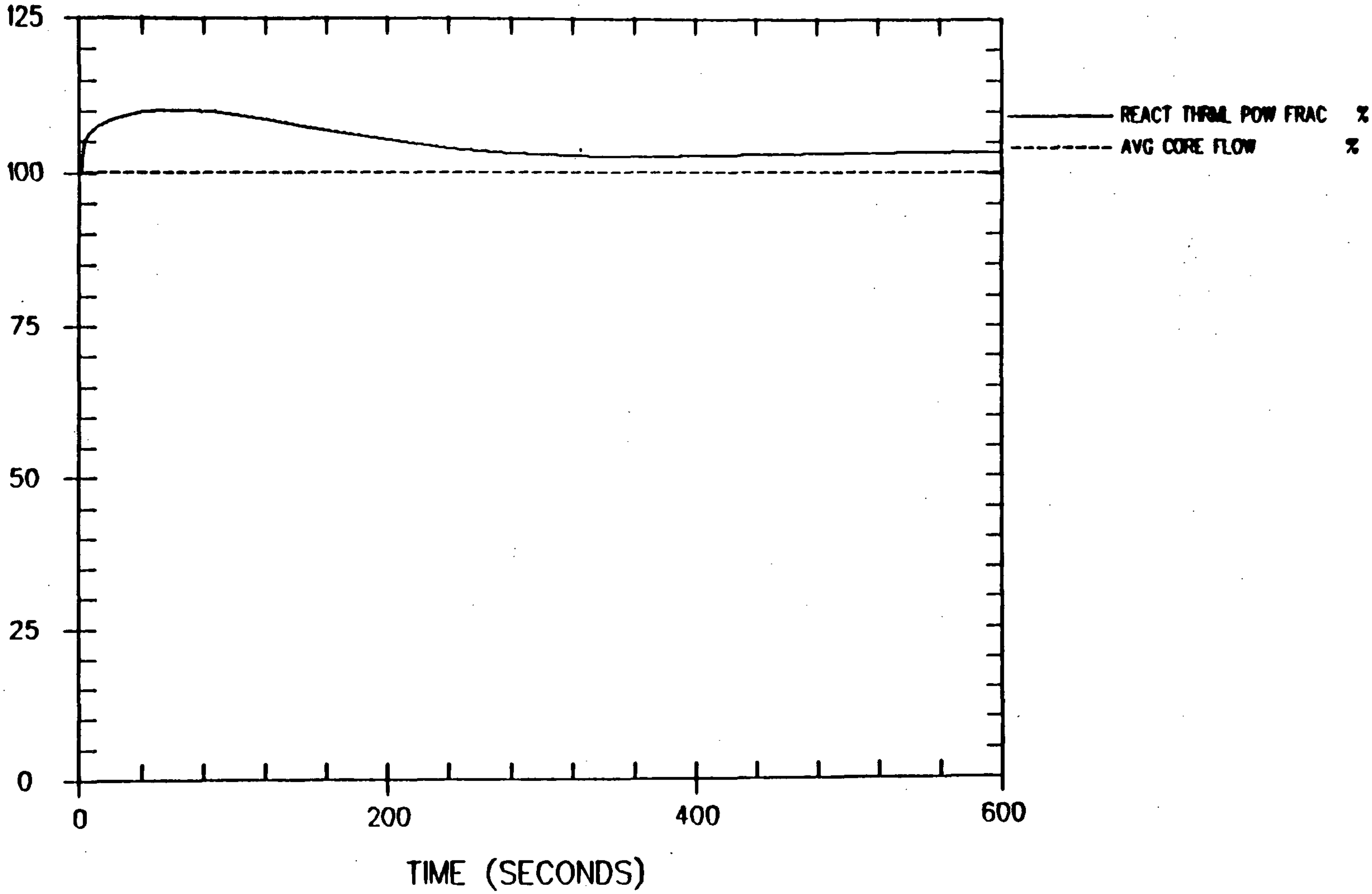


FIGURE E.6-3a REACTOR POWER AND PRIMARY FLOW FOLLOWING INADVERTENT WITHDRAWAL OF A CONTROL ROD, COMBINED WITH FAILURE TO SCRAM (UTOP)

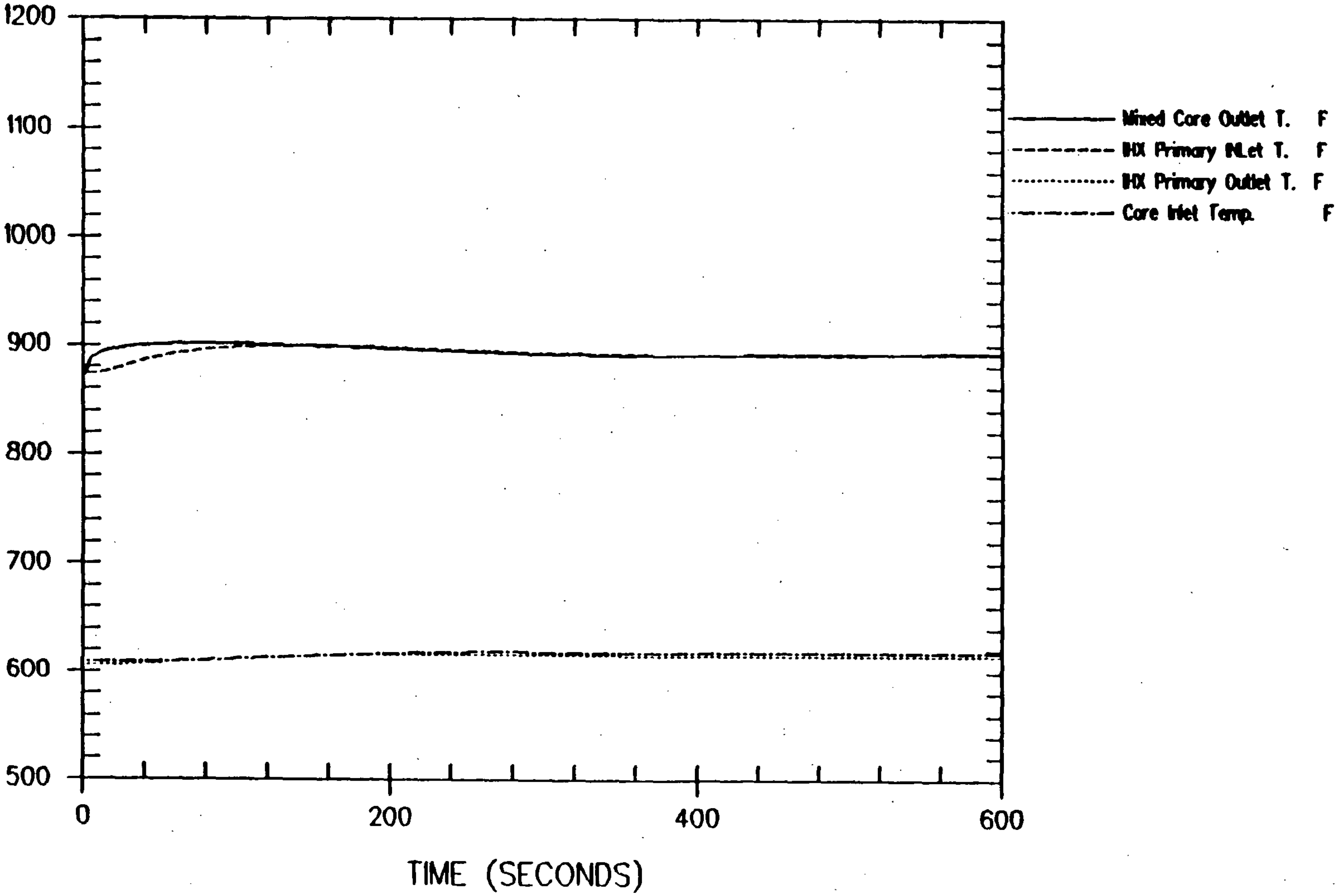


FIGURE E.6-3b PRIMARY SODIUM TEMPERATURES FOLLOWING INADVERTENT WITHDRAWAL OF A CONTROL ROD, COMBINED WITH FAILURE TO SCRAM (UTOP)

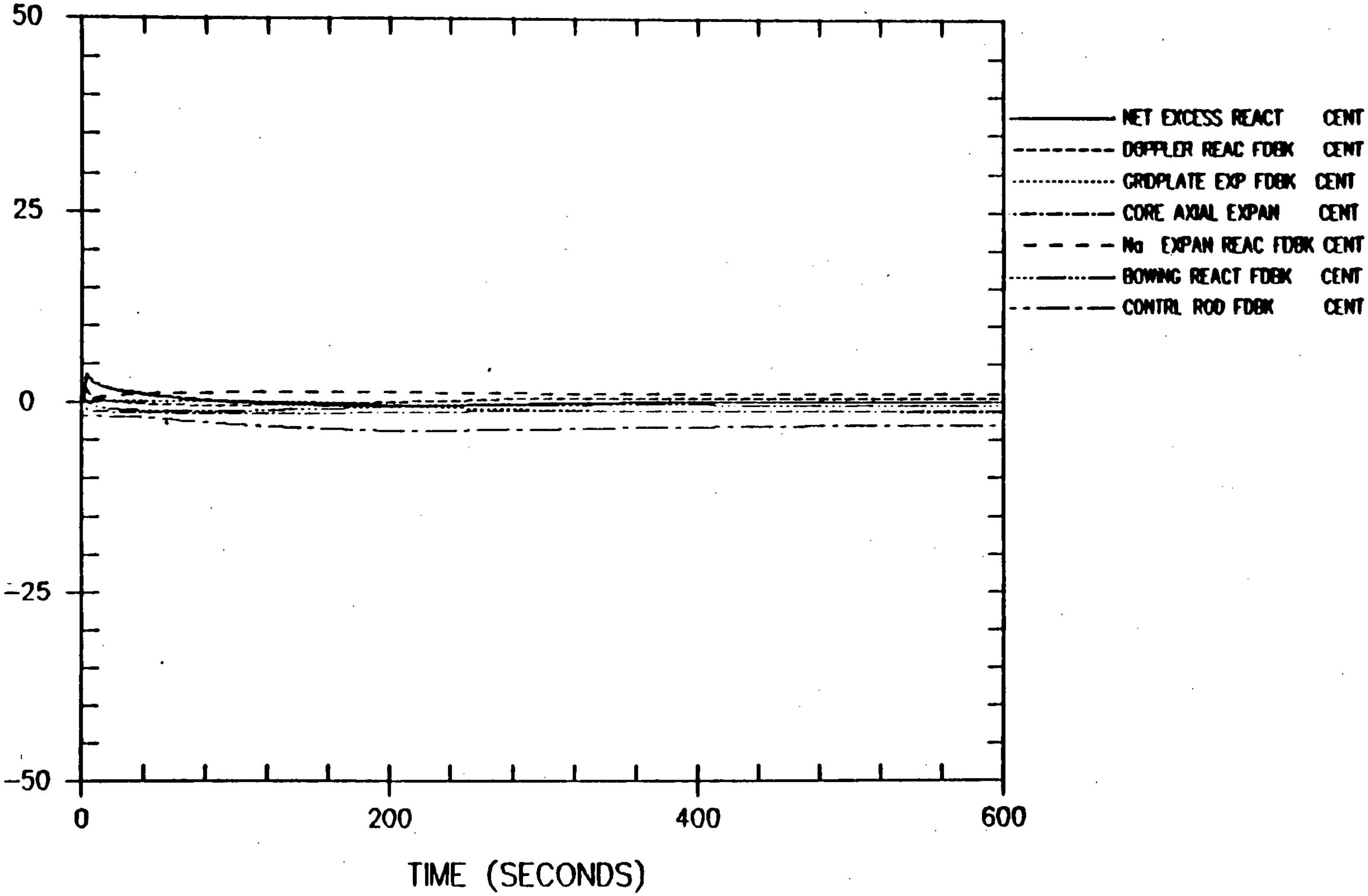


FIGURE E.6-3c CORE INHERENT REACTIVITY FEEDBACK FOLLOWING INADVERTENT WITHDRAWAL OF A CONTROL ROD, COMBINED WITH FAILURE TO SCRAM (UTOP)

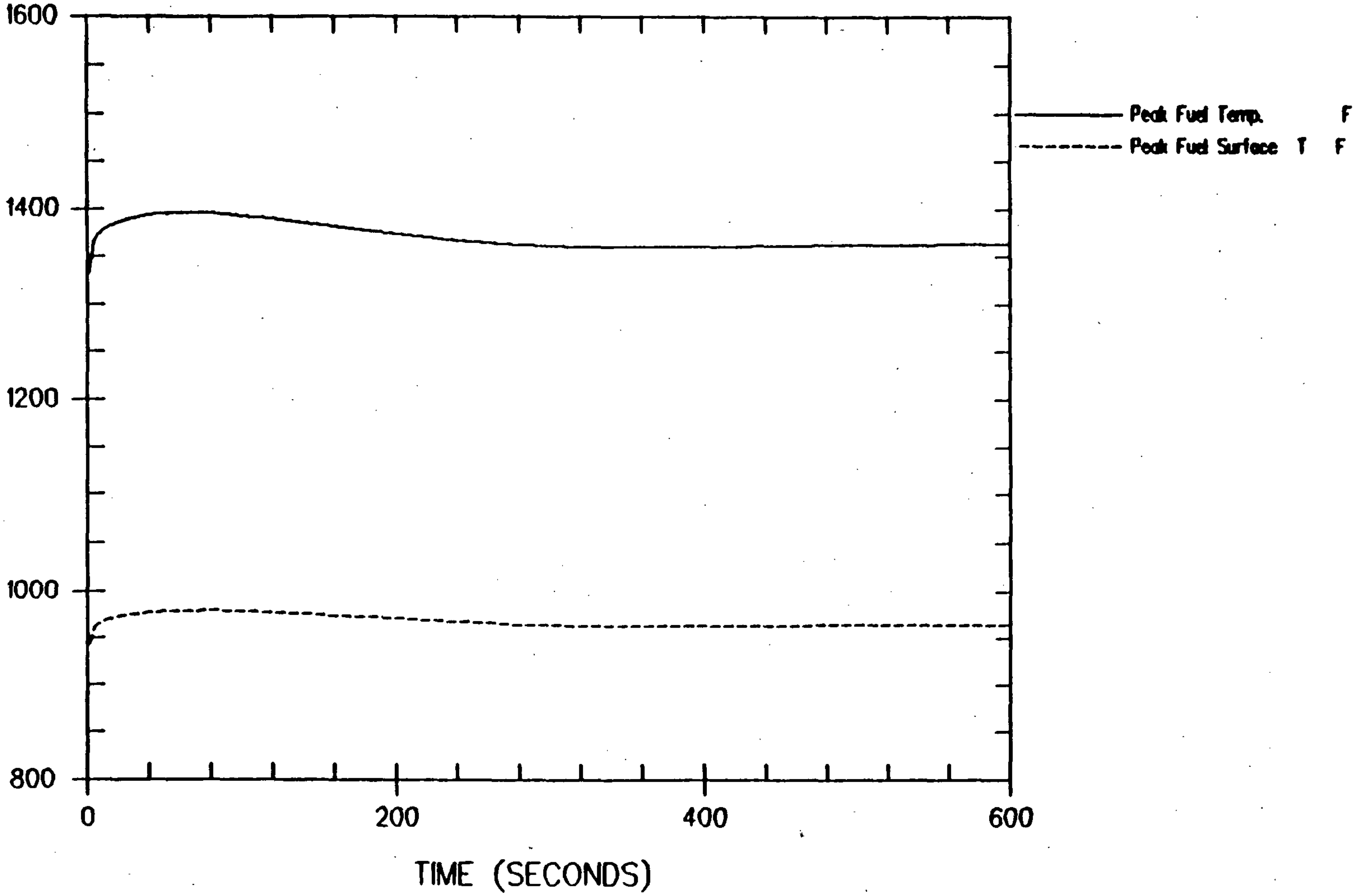


FIGURE E.6-3d PEAK FUEL CENTERLINE AND SURFACE TEMPERATURES FOLLOWING INADVERTENT WITHDRAWAL OF A CONTROL ROD, COMBINED WITH FAILURE TO SCRAM (UTOP)

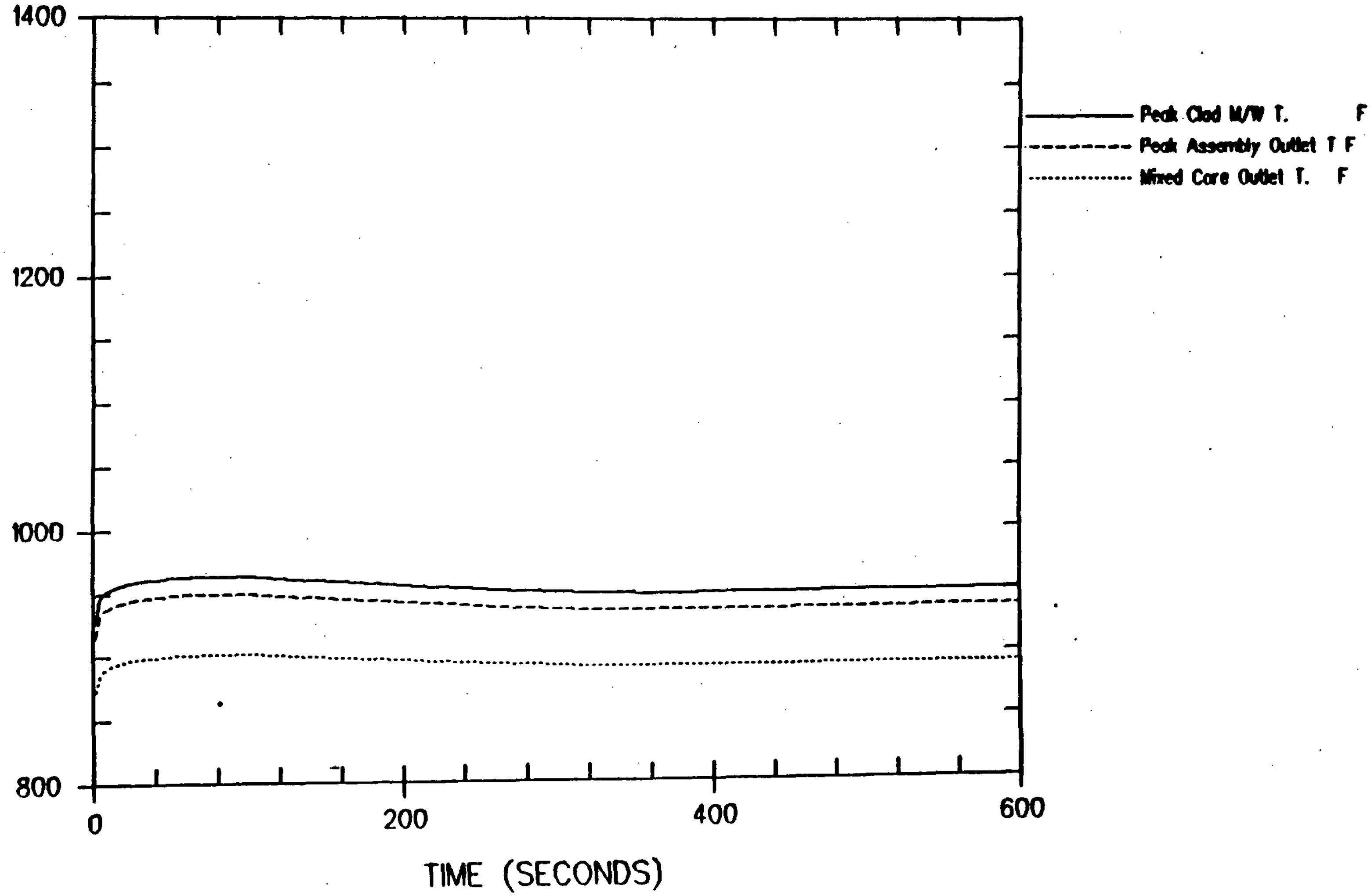


FIGURE E.6-3e PEAK CLADDING AND SODIUM TEMPERATURES AND MIXED CORE OUTLET TEMPERATURES FOLLOWING INADVERTENT WITHDRAWAL OF A CONTROL ROD, COMBINED WITH FAILURE TO SCRAM (UTOP)

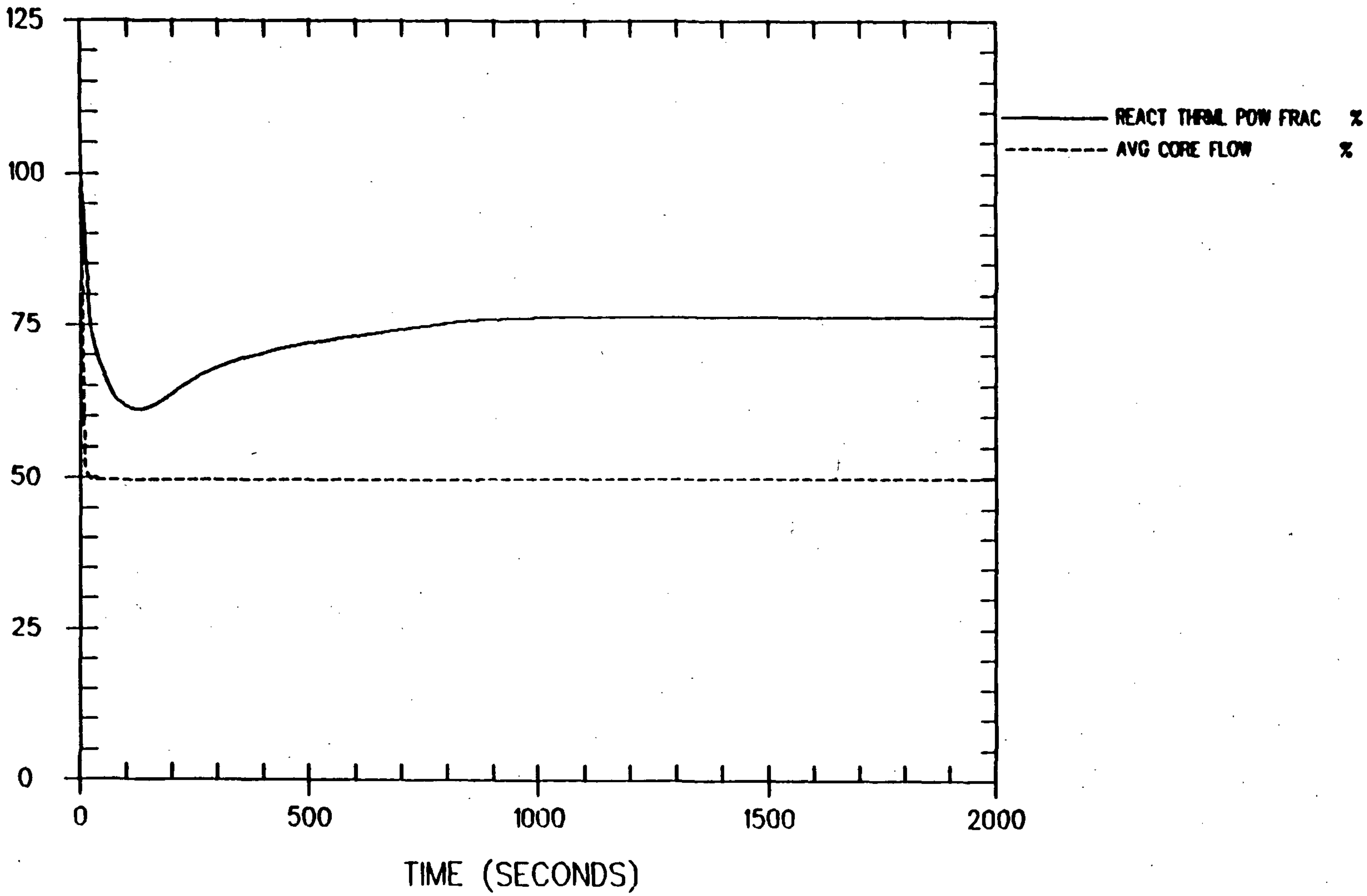


FIGURE E.6-4a REACTOR POWER AND PRIMARY FLOW FOLLOWING TRIP OF A PRIMARY PUMP COMBINED WITH FAILURE TO SCRAM (ULOPP)

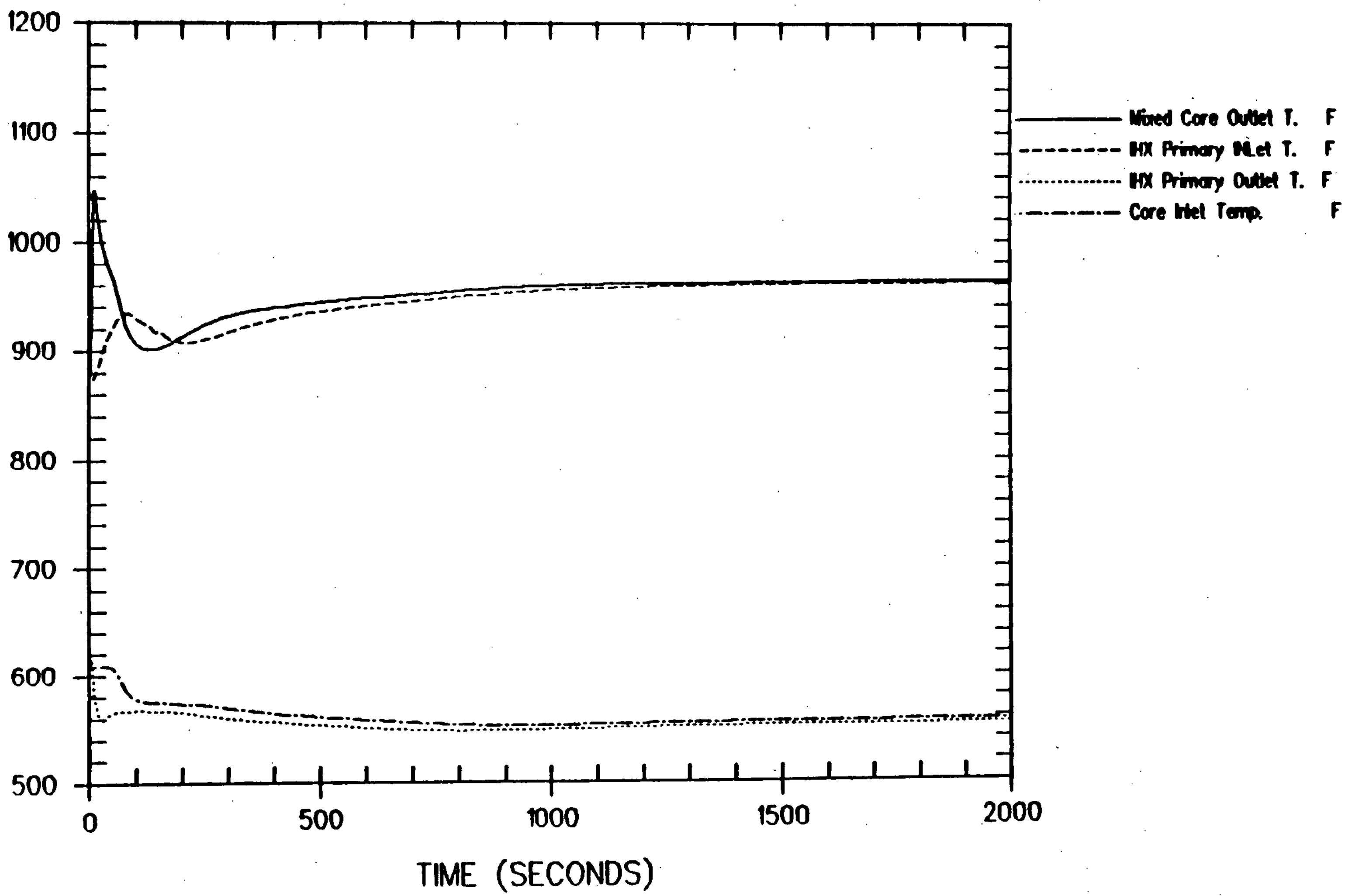


FIGURE E.6-4b PRIMARY SODIUM TEMPERATURES FOLLOWING TRIP OF A PRIMARY PUMP COMBINED WITH FAILURE TO SCRAM (ULOPP)

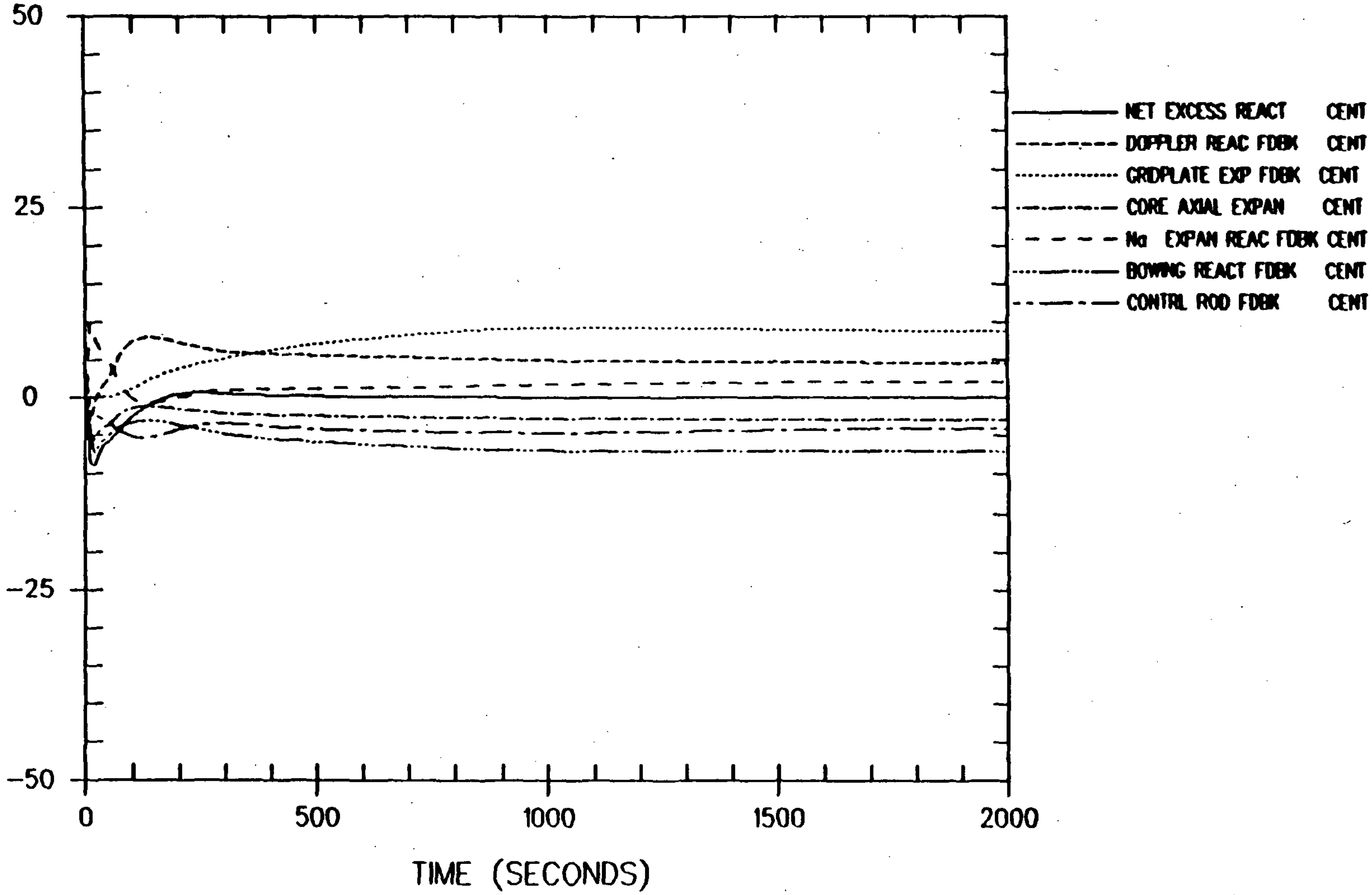


FIGURE E.6-4c CORE INHERENT REACTIVITY FEEDBACK FOLLOWING TRIP OF A PRIMARY PUMP COMBINED WITH FAILURE TO SCRAM (ULOPP)

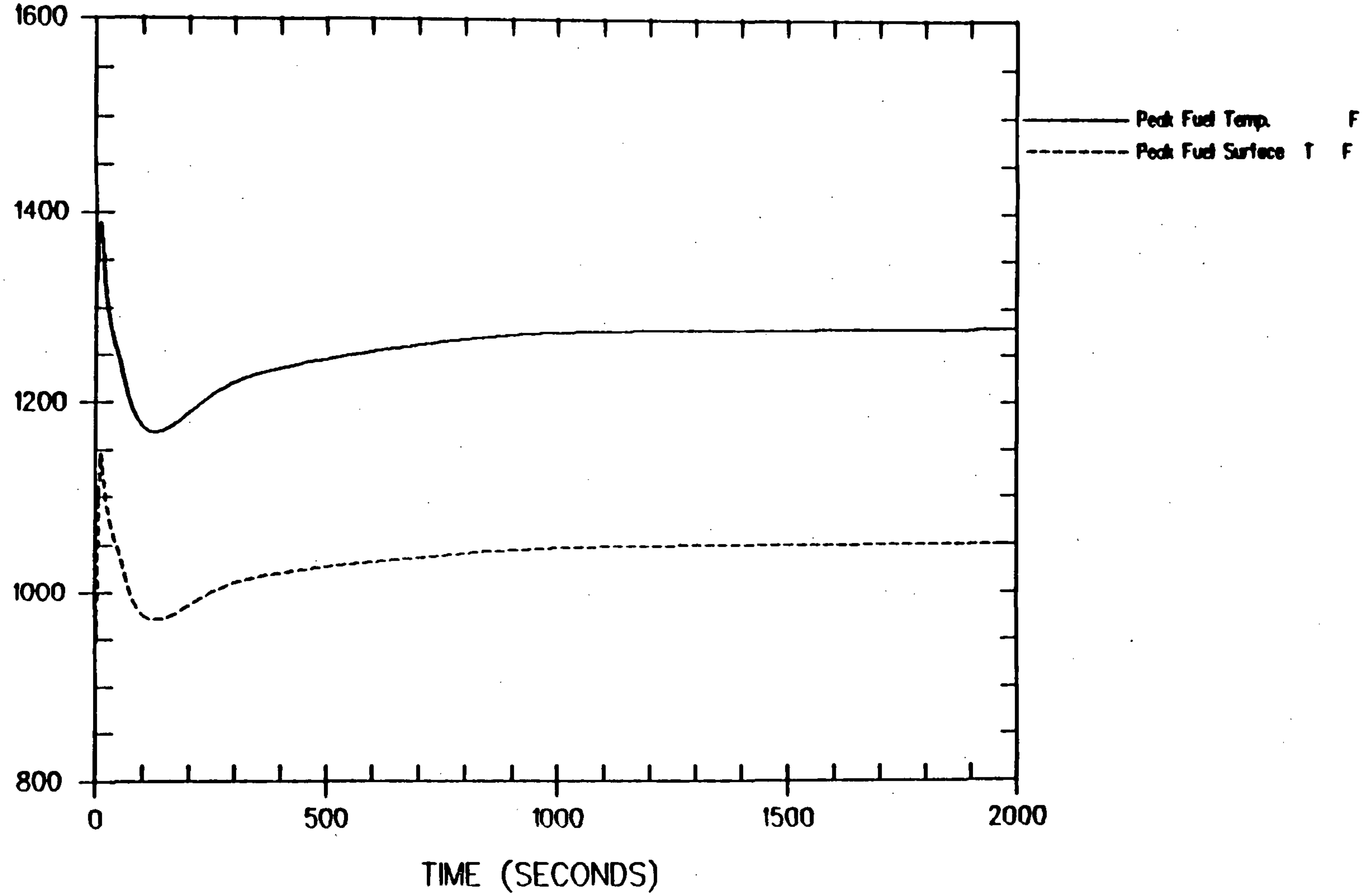


FIGURE E.6-4d PEAK FUEL CENTERLINE AND SURFACE TEMPERATURES FOLLOWING TRIP OF A PRIMARY PUMP COMBINED WITH FAILURE TO SCRAM (ULOPP)

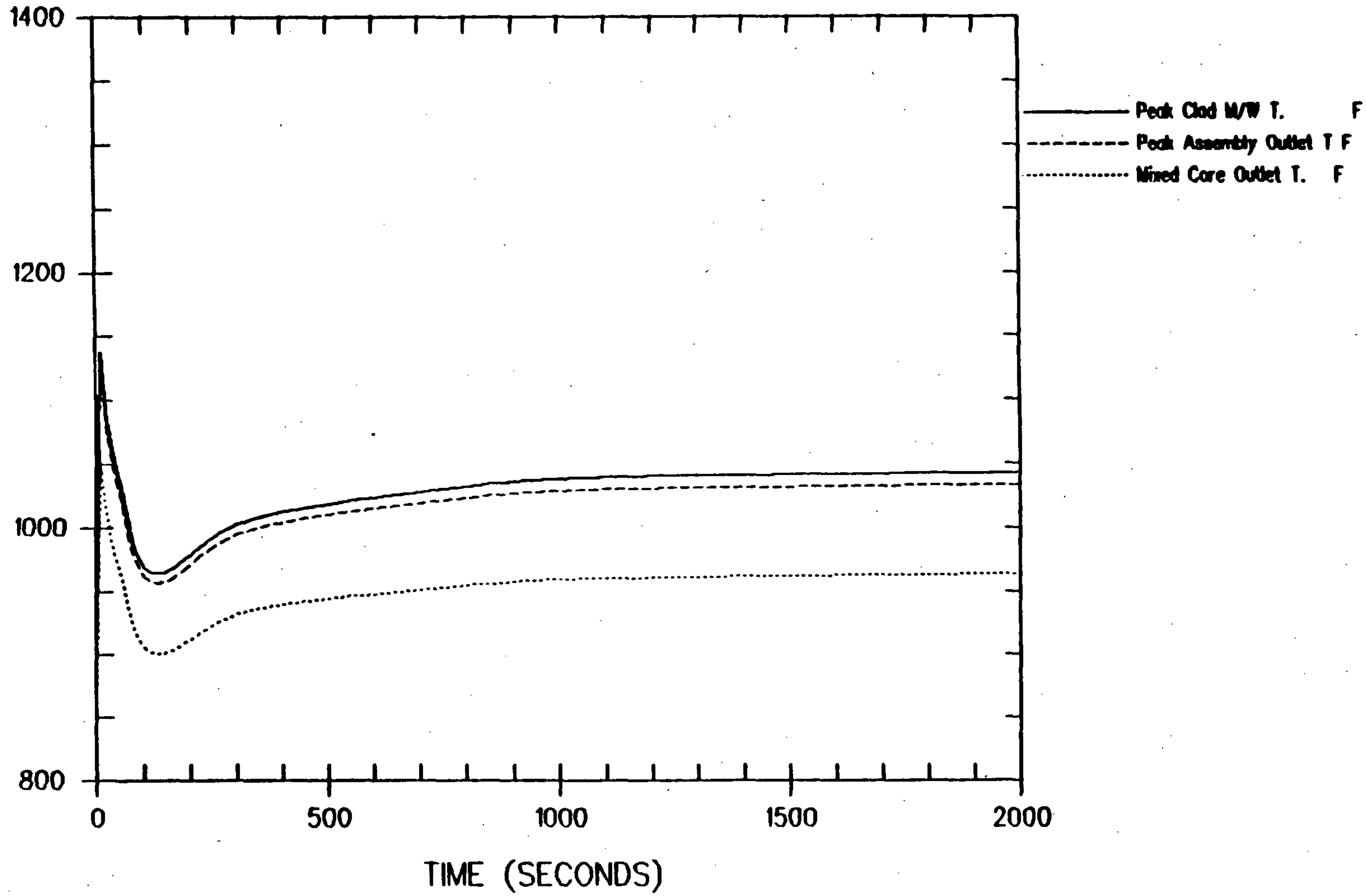


FIGURE E.6-4e PEAK CLADDING AND SODIUM TEMPERATURES AND MIXED CORE OUTLET TEMPERATURE FOLLOWING TRIP OF A PRIMARY PUMP COMBINED WITH FAILURE TO SCRAM (ULOPP)

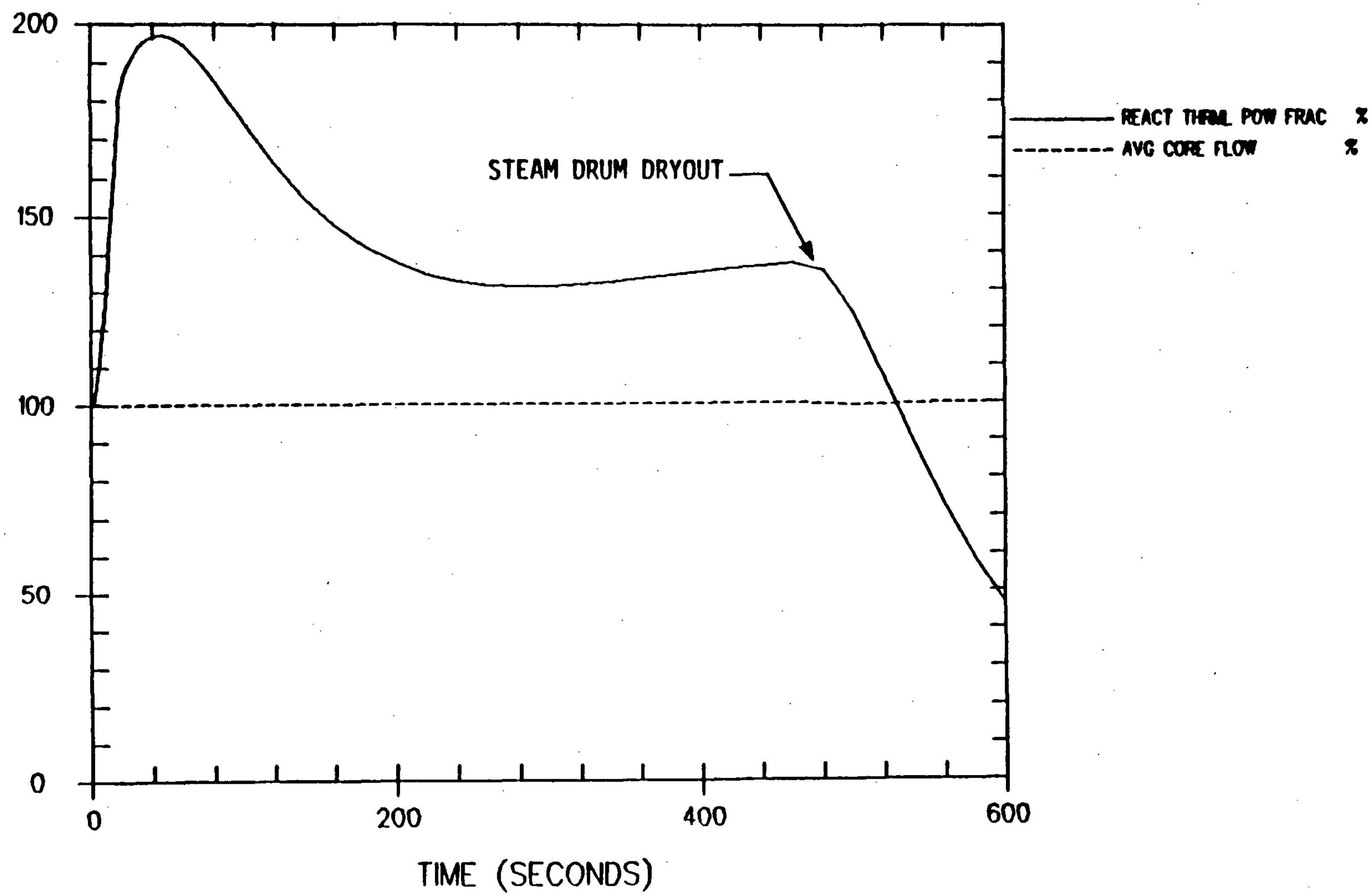


FIGURE E.6-5a REACTOR POWER AND PRIMARY FLOW FOLLOWING ALL-RODS TRANSIENT OVERPOWER, COMBINED WITH FAILURE TO SCRAM (UTOP-6 RODS)

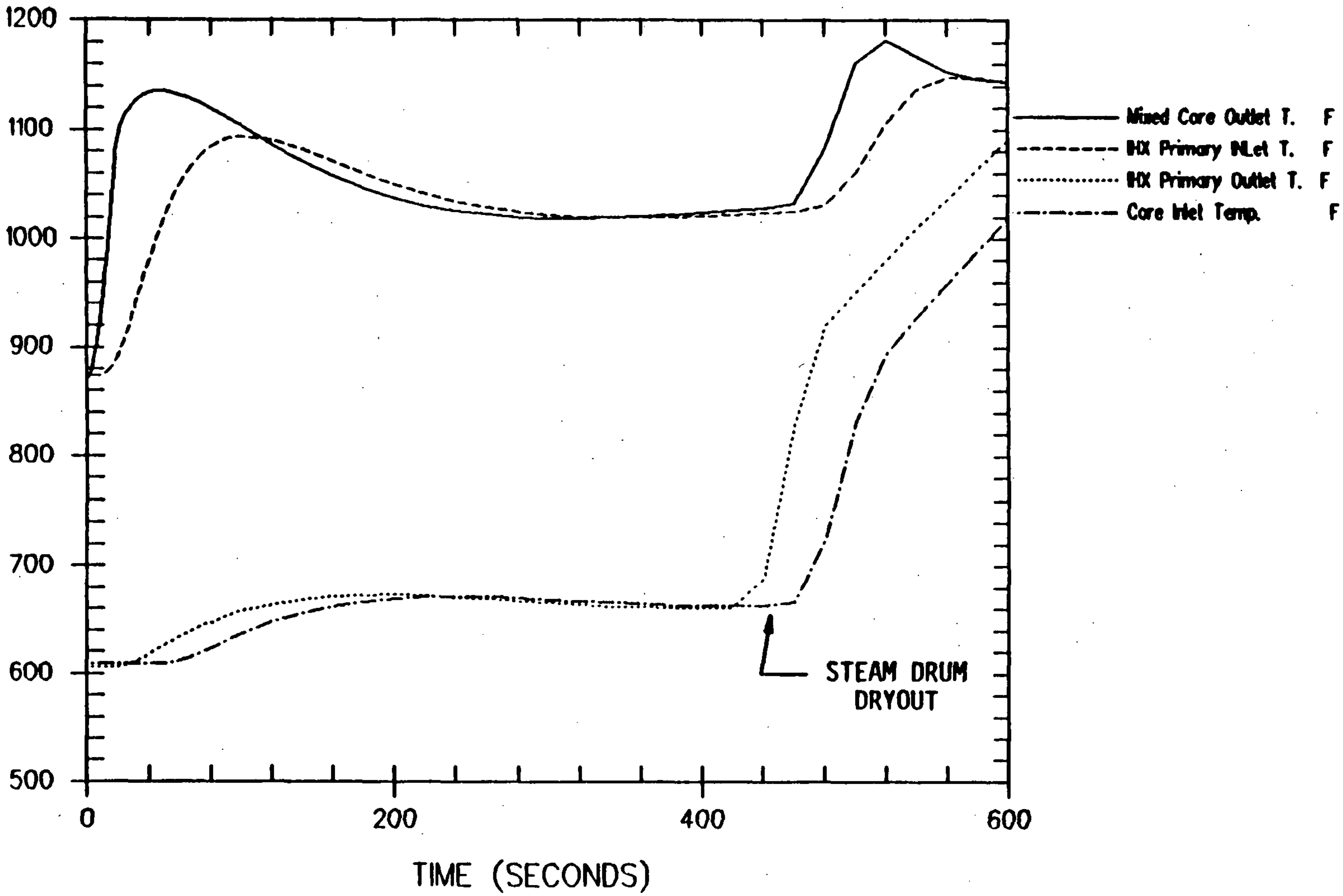


FIGURE E.6-5b PRIMARY SODIUM TEMPERATURES FOLLOWING ALL-RODS TRANSIENT OVERPOWER, COMBINED WITH FAILURE TO SCRAM (UTOP-6 RODS)

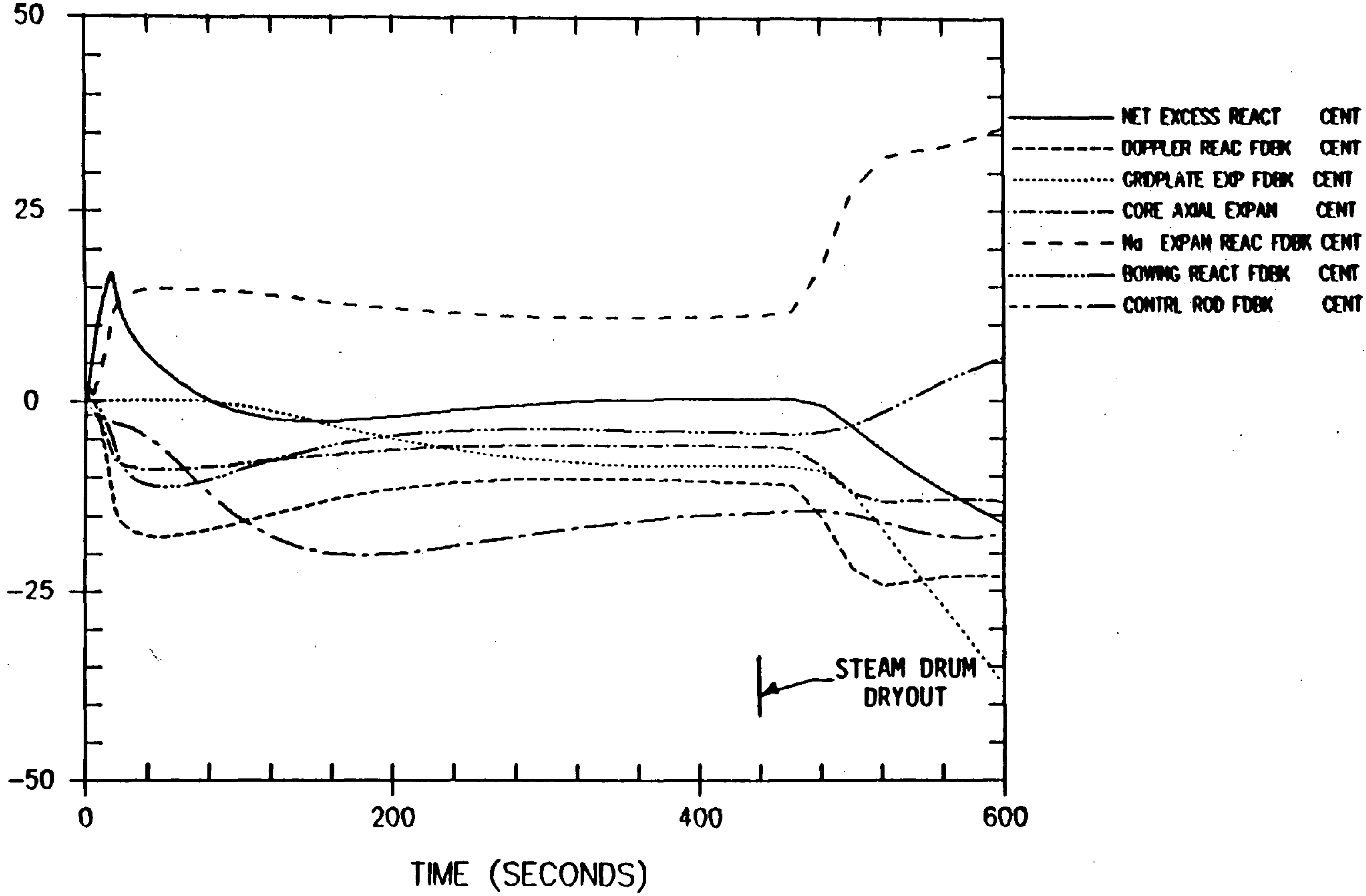


FIGURE E.6-5c CORE INHERENT REACTIVITY FEEDBACK FOLLOWING ALL-RODS TRANSIENT OVERPOWER, COMBINED WITH FAILURE TO SCRAM (UTOP-6 RODS)

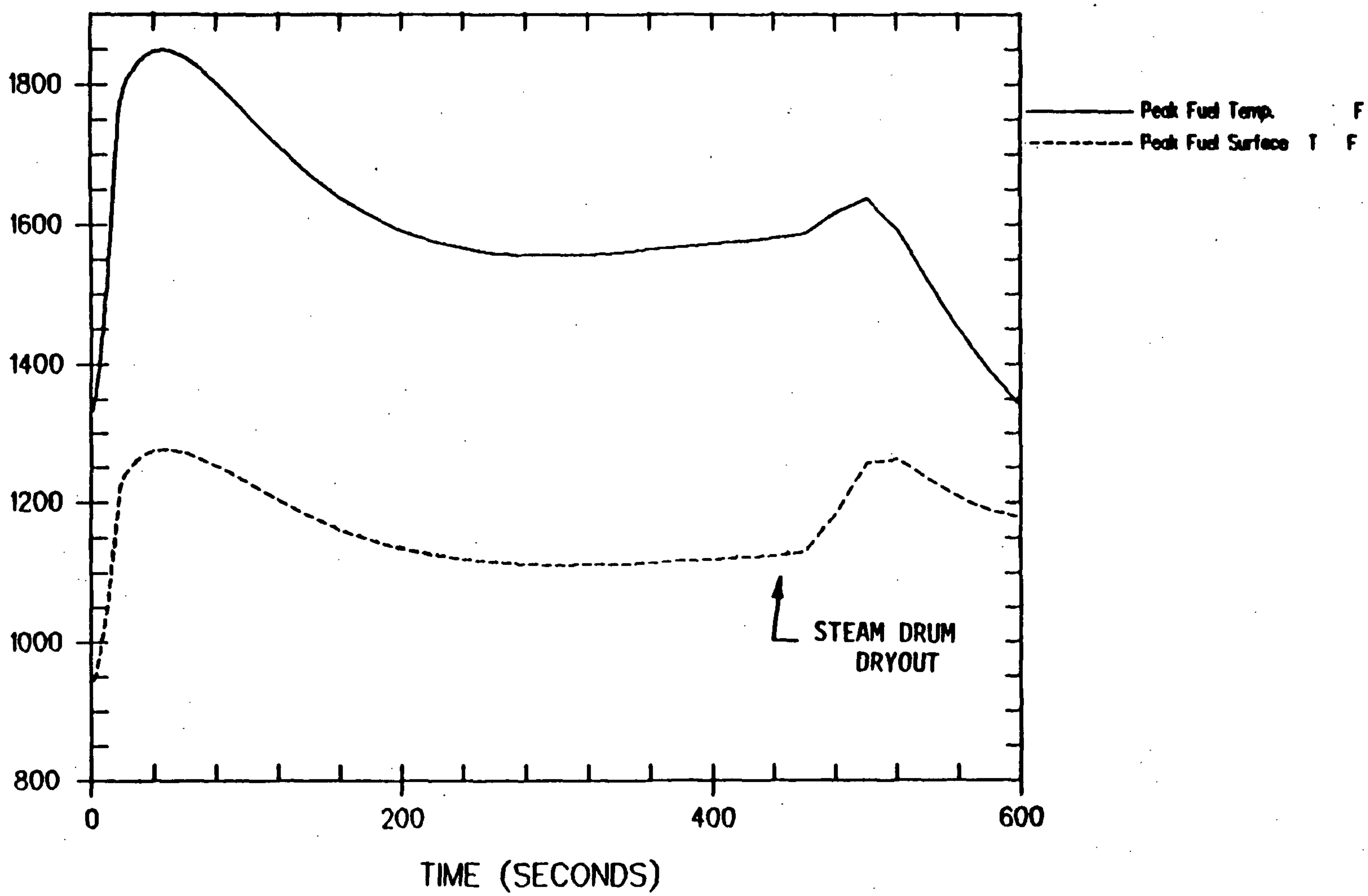


FIGURE E.6-5d PEAK FUEL CENTERLINE AND SURFACE TEMPERATURES FOLLOWING ALL-RODS TRANSIENT OVERPOWER, COMBINED WITH FAILURE TO SCRAM (UTOP-6 RODS)

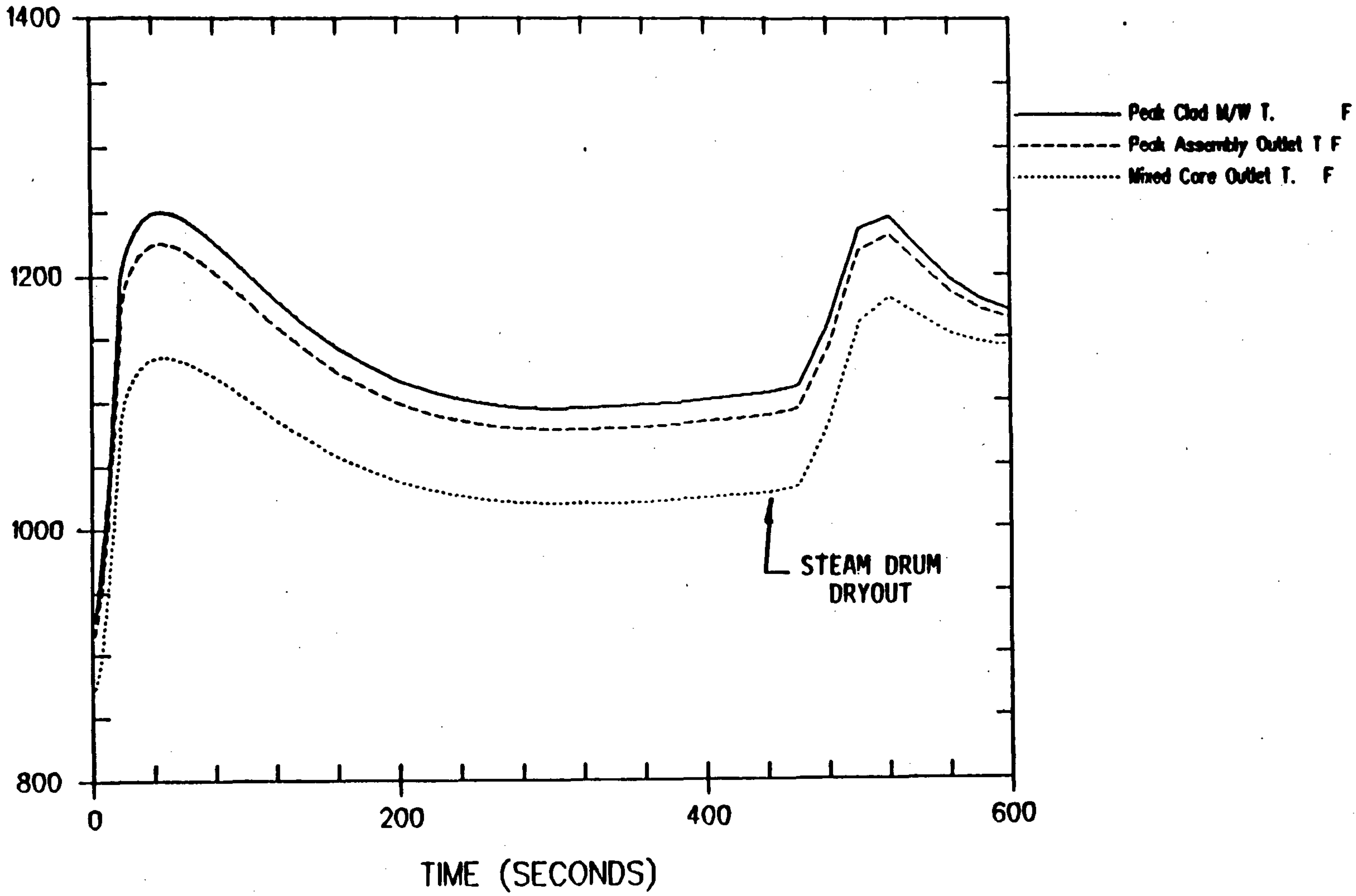


FIGURE E.6-5e PEAK CLADDING AND SODIUM TEMPERATURES AND MIXED CORE OUTLET TEMPERATURES FOLLOWING ALL-RODS TRANSIENT OVERPOWER, COMBINED WITH FAILURE TO SCRAM (UTOP-6 RODS)

E.8 Comment

Bounding Events (BE) have been the subject of discussions during the past several months.

Provide analyses of the PRISM conceptual design for the latest list of the Bounding Events shown below. The analyses should include assessments of the thyroid and whole body doses at the site boundary for 36 hours and 30 days.

Deterministic Events

These events are intended to bound the LMR DBA and BDBA spectrum in order to account for PRA uncertainties and provide conservatism in selecting a SSST and assessing the adequacy of containment and off-site evacuation plans. These events are judged to be bounding for the following categories.

- Reactivity insertion
- Heat removal
- Loss of coolant
- Na/H_2O reaction

Event 1: Inadvertent withdrawal of all control rods without scram for 36 hours (single module):

- Forced cooling
- RACS/RVACS cooling only

Event 2: Station blackout for 16 hours:

- Loss of all ac power

Event 3: Loss of forced cooling plus DRACS/RACS/RVACS with scram (single module):

- 25% partial unblockage after 36 hours

Event 4: Instantaneous loss of flow from one primary pump (single module):

- Coastdown flow for other pump
- Without scram for 36 hours

Event 5: Steam generator tube rupture with failure to isolate or dump water from steam generator:

- Justifiable number of tube failures
- Defined sequence of ruptures

Event 6: Large Na leaks (single module):

- Double ended guillotine break of IHTS pipe
- RV leak (critical leak)

Event 7: External events consistent with their treatment for LWRs.

E.8 Response

Because this response is complex and lengthy, it is divided into several sections:

A. General Comments on the Proposed Deterministic Events
B. Specific Comments and Probability Estimates
C. Analysis of the Events
D. Radiological Dose Assessments

Addendum A - Tolerance of RVACS to Blockages

RESPONSES TO NRC COMMENTS

A. General Comments on the Proposed Deterministic Events.

Based on our previous discussions with the NRC Staff and our study of the presentation to the ACRS dated February 11, 1988, regarding Advanced Reactors by Mr. T. L. King, NRC, we understand that the use of the Bounding Deterministic Events is meant to accomplish a number of purposes. Namely:

- o test the design against severe accidents, including anticipated transients without scram,
- o explore the design for thresholds such as major fuel damage, sodium boiling, and structural damage,
- o test the design for long response times under severe accidents and determine the need for off-site evacuation planning,
- o provide a basis for the selection of a mechanistic site suitability source term, and
- o test the adequacy of containment and meeting 10CFR100 requirements.

It appears to us that using a single set of events for these diverse purposes is not appropriate, although it may be useful as a first step in initiating the evaluation process. For example, the events forming the mechanistic basis for the site suitability source term might include events resulting in substantial fuel failures and a challenge to the containment, even if those events were of very low probability, in order to provide an appropriate test of the containment function against a substantial radiation release from the core. The same set of events would probably not be appropriate to establish the sufficiently low probability of even a limited radiation release to the environment for the consideration of the requirement for off-site evacuation planning.

It should be noted that some of the events have implied provisions. For example, only very special designs can meet event 1, withdrawal of all control rods without scram, without major damage; specifically, designs with very low excess core reactivity. We do not believe that other designs, such as a liquid metal cooled reactor with an oxide fuel core, are intended to be ruled out. Therefore we understand that there is the implied provision associated with this event: provided that there is no safety-grade, electronic rod-block feature in the reactor protection system, or a safety-grade mechanical rod stop in the drive mechanism, this event should be considered. It appears that some of the other events listed also have similar implied provisions, which remain open to further discussion and definition.

Some of the events listed are, by our analyses, of such low probability for PRISM that they belong in the realm of the probabilistic risk assessment (PRA). An example is event 3, loss of all decay heat removal, even by passive systems for 36 hours. The evaluation of this event may be useful to determine the time available before reaching thresholds of fuel damage, sodium boiling, and containment pressure and temperature limits under extreme limiting conditions. In addition to that evaluation, however, we intend to show that the passive decay heat removal system characteristics of PRISM are such that even severe blockage in any region, or even complete blockage in some key regions, does not result in complete loss of cooling. Thus, the passive decay heat removal system can accommodate severe structural failures and ingestion of foreign materials and still retain adequate cooling capability.

From the inception of the PRISM design, the intention always was to comply with the NRC safety goals. To this end, the design was carried out to meet ASME limits and to avoid significant fuel failures for all internal events down to a probability of 10^{-6} per year and for an additional set of anticipated transient without scram (ATWS) events with probabilities below 10^{-6} per year. The probabilistic risk assessment (PRA) performed on the design indicates that the NRC safety goals are met with substantial margin even without relying on evacuation. We believe that the existing design is fully responsive to both the NRC safety goals and to the NRC Advanced Reactors policy which calls for design characteristics such as inherent and passive features, long response times, reduced potential for operator errors, etc., with the overall goal of enhanced safety margins.

We are concerned that in implementing the NRC safety goals, the NRC Staff is developing additional criteria which unduly compound required margins. The NRC's basic safety goals call for the risk to a member of the public in the vicinity of a power plant to be no more than 0.1 percent of the risk of prompt fatality from other accidents or cancer from other sources to which he is already exposed. This goal is equivalent to a 0.5×10^{-6} per year probability of prompt fatality and a 2×10^{-6} per year probability of cancer to an individual in the vicinity of the power plant. In implementing these already conservative goals, the NRC Staff is calling for the probability of a large radiation release to be below 10^{-6} per reactor year. It is widely recognized that the latter figure is much more restrictive than the safety goals; see for example the article by Whipple and Starr, Nuclear Power Safety Goals in Light of the Chernobyl Accident; Nuclear Safety, Vol. 29, No. 1, January-March 1988. It is clearly lower than the cancer risk goal, and the only way it could be consistent with the fatality risk goal is if, in the event of a large radiation release, the probability of fatality to an individual were 50 percent. Consideration of wind direction and population distribution result in a probability which is at least an order of magnitude less.

A further step in the compounding of margins is the interpretation of large radiation release to mean 25 Rem exposure at the boundary or near vicinity of the plant.

A further step is the addition of the prescribed Bounding Events to those evaluated based on the 10^{-6} per year release criterion plus ATWS events, some of which, as we indicated earlier, are of such low probability that they belong in the PRA analyses.

An additional compounding step is represented by the proposed requirement that in order to eliminate off-site evacuation planning, the probability be less than 10^{-6} per year that a 1 REM exposure occurs at the site boundary during 36 hours after the beginning of an accident event, and that the same exposure limit be met for the Bounding Events. This in effect calls for the probability of a limited radiation release to be below 10^{-6} per year, even though the probabilistic risk assessment indicates that the safety goals are met with substantial margins with no evacuation, without the imposition of this additional requirement.

For the reasons described above, we regard the list of bounding events as a first step in achieving the various purposes mentioned at the beginning of this discussion. The analyses provided in the subsequent sections indicate the capabilities of the PRISM design under severe events and the long response times before failure thresholds are reached. We request that the NRC Staff reevaluate the selection and use of the list of Bounding Events taking into account the foregoing discussion, and the probabilities and analyses provided in this response.

B. Specific Comments And Probability Estimates For The Events In The PRISM Design

Event 1 Inadvertent withdrawal of all control rods without scram for 36 hours (single module):

a. forced cooling

b. RACS/RVACS cooling only

Comments on Event 1a

This event was previously analyzed as an ATWS event, in view of the fact that the plant control system is not safety grade and that a safety grade rod-block feature is not provided. If the consequences of this event are found to be too severe, either safety grade electronic rod-blocks or mechanical rod stops should be acceptable to reduce the reactivity addition by the control rods.

In the analysis of this event, it was conservatively assumed that the steam generator would dry out after 8 minutes, and that at that time all cooling through the intermediate heat transport systems (IHTS) would stop. An updated analysis is provided.

Probability of Event 1a

Plant control system failure, causing

Withdrawal of all control rods	10^{-2}/year
Plant control system fails to stop rod withdrawal and to initiate rod runback based on independent power or temperature signal	10^{-2}/demand
Scram system fails to insert any one rod, given that all rods were moving (i.e., they are not stuck)	10^{-7}/demand
Subtotal, initial event probability, one module	10^{-11}/year
Subtotal, 3 module power block	3×10^{-11}/year
No plant control system recovery for 36 hours	$<10^{-1}$
No scram system recovery for 36 hours	$<10^{-2}$
Total long term event probability	$<3 \times 10^{-14}$/year

Comment on Event 1b

As noted above, in the analysis of event 1 a., it was already assumed that all heat removal through the IHTS would stop after 8 minutes. This event adds the coincident failure of the IHTS during the first few minutes of event 1 a. The analysis of this event is provided; however, we consider this event of such low probability that it belongs in the probabilistic risk assessment (PRA).

RESPONSES TO NRC COMMENTS

<u>Probability of Event 1b</u> - Initiated by Control System Failure

Event 1a initial event probability for 3 module power block	3×10^{-11}/year
Simultaneous failure of the IHTS	$<10^{-4}$
Initial event probability, 3 module power block	$<3 \times 10^{-15}$/year

<u>Probability of Event 1b</u> - Initiated by IHTS Failure

Complete IHTS failure	$<10^{-4}$/year
Plant control system failure, causing withdrawal of all control rods	$<10^{-2}$/demand
Plant control system fails to stop rod withdrawal and to initiate rod runback based on independent power or temperature signal	$<10^{-2}$/demand
Scram system fails to insert any one rod, given that all rods were moving (i.e., they are not stuck)	$<10^{-7}$/demand
Initial event probability, one module	$<10^{-15}$/year
Initial event probability, 3 module power block	$<3 \times 10^{-15}$/year

Event 2 Station blackout for 16 hours.
- loss of all ac power

<u>Comment on Event 2</u>

This event is bounded by the ATWS event analyzed previously which assumes loss of primary flow and loss of IHTS cooling, without scram. An updated analysis is provided.

<u>Probability of Event 2</u>

Loss of off-site power from preferred supply	0.2/year
Unavailability of power from secondary off-site supply	10^{-2}
Failure of turbine generator to runback to house load (i.e., generator trip)	10^{-3}/demand
Subtotal initial event probability for 3 module power block (disregarding gas turbine backup or power from other power blocks)	2×10^{-6}/year
No power recovery for 16 hours	$<10^{-3}$
Total long term event probability for 3 module power block	$<2 \times 10^{-9}$/year
Multiplier for failure to scram case	
Failure to scram on demand, 1 module	3×10^{-7}/demand
Failure to scram any of 3 modules in a 3-module power block	9×10^{-7}

Event 3 Loss of forced cooling plus DRACS/RACS/RVACS with scram (single module):
- 25% partial unblockage after 36 hours

Comments on Event 3

This event calls for total loss of decay heat removal by all systems, including that by the passive, natural circulation reactor vessel auxiliary cooling system (RVACS). Since previous discussions of this event with the NRC staff, we have made further analyses of the tolerance of RVACS to blockages. These analyses indicate that severe blockage, and even complete blockage of the air path at certain points, does not result in loss of decay heat removal. For example, complete blockage of all air inlets, or complete blockage of the air turnaround path at the bottom of the silo by a postulated ingestion of material, would result in only an approximately 50°F increase in the maximum primary sodium temperature. This is because natural air circulation will be established within the air ducts which are normally the exhaust ducts; some of the air outlet stacks become inlet stacks. Structural failures or collapse of the inlet and exhaust stacks blocking 90% of the air passages would result in less than 100°F increase in the maximum sodium temperature. The analysis results of such severely degraded cases are provided.

Our opinion is that the specified 36 hours time period before unblocking is much longer than necessary. We believe that the only way complete blocking can occur is by all RVACS inlets and outlets being covered in some way at their above grade locations. This kind of blockage can, we believe, be substantially removed in no more than 8 hours. All the other methods of blocking of RVACS, including complete structural collapse and water or sand flooding still leave adequate air passages to allow enough cooling to stay below acceptable temperature limits.

Our analyses indicate that the dominant internal initiator resulting in total loss of decay heat removal is the sodium leak through both the reactor vessel and containment vessels; the other potential causes being extreme external events and sabotage. The probabilities are such that all these events belong in the realm of the PRA. Our understanding is that the NRC staff is not asking to consider the double vessel leak (see Event 6), and that external events will be specified later under Event 7. For the hypothetical case where saboteurs have succeeded in blocking all inlets and outlets of four RVACS stacks, the analyses provided show the time periods available to effect partial unblocking before fuel failures, sodium boiling or structural failures occur.

Probability of Event 3 - Caused by Double Vessel Leak

Leak in containment vessel or reactor vessel	2×10^{-6}/years
Leak in other vessel before fuel is unloaded	5×10^{-7}
Total event probability, 1 module	10^{-12}/year
Total event probability, 3 module power block	3×10^{-12}/year

Event 4 Instantaneous loss of flow from one primary pump (single module):
- **coast down flow for other pump**
- **without scram for 36 hours**

Comments on Event 4

The loss of primary flow without scram has been considered before as an ATWS event. Event 4 adds the failure of one pump to coast down. For the sake of simplicity, the analysis provided is the same as for Event 2, but with failure of one pump to coast down.

Probability of Event 4

Instantaneous loss of flow from one pump (EM pump shorts out)	10^{-2}/year
Failure to scram	3×10^{-7}/demand
Simultaneous trip of other 3 pumps (failure of RPS to delay pump trip)	10^{-3}/demand
Subtotal, initial event probability, 1 module	3×10^{-12}/year
Subtotal, 3 module power block	9×10^{-12}/year
No scram recovery for 36 hours	$<10^{-2}$
Total long term event probability, 3-module power block	$<9 \times 10^{-14}$/year

Event 5 Steam generator tube rupture with failure to isolate or dump water from steam generator:
- **justifiable number of tube failures**
- **defined sequence of ruptures**

Comment on Event 5

Analysis and extensive discussion of this event is provided; however, we consider failure to isolate and blow down for an extended period of time of such low probability that it belongs in the Probabilistic Risk Assessment (PRA).

Probability of Event 5

Steam generator tube leak (per unit)	10^{-3}/year
Failure of operator actions to isolate and blowdown before bursting of rupture disk	10^{-1}/event
Failure of automatic isolation and blowdown system	3×10^{-5}/demand
Subtotal, initial event probability, 1 module	3×10^{-9}/year
Subtotal, 3-module power block	10^{-8}/year
Failure to trip turbine and isolate other two modules shutting off steam backflow and feedwater	10^{-2}/event
Total event probability, 3-module power block	10^{-10}/year

Event 6 Large Na leaks (single module):
- **Double ended guillotine break of IHTS pipe**
- **RV leak (critical leak)**

Comments on Event 6

Because of the low stress, low energy nature of the IHTS, a leak-before-break situation is expected to exist for the pipes.

Reactor vessel leak (at any location) has been taken as a design basis event for PRISM.

Analyses of these events are provided in a subsequent section.

Probabilities for Event 6

Double ended guillotine break of IHTS pipe	10^{-8}/year
Reactor vessel leak	10^{-6}/year

Event 7 External events consistent with their treatment for LWRs

We understand that these events will be specified later by the NRC staff as part of the implementation of the NRC policy on severe accidents.

C. Analysis of Events

The first four events on the NRC list have been analyzed using the GE ARIES plant transient analysis code. This code is very similar to, and has been validated against, the national LMR transient safety analysis code SASSYS. In each case, a nominal analysis has been performed; specifically, the expected magnitude of assembly duct bending as predicted by ANL NUBOW-3D analysis of the reference PRISM core has been included.

Damage/Failure Criteria

The relevant damage/failure criteria are the following:

1. Cladding Failure by Fuel-Clad Eutectic Attack - Cladding rupture from weakening by internal wastage by fuel-clad eutectic attack is the principal fuel pin failure phenomenon. Figure E.8-1 relates the internal wastage rate to the temperature at the fuel-clad interface, where a low melting temperature alloy forms from the fuel and cladding constituents. At temperatures below the alloy melting temperature of 1290°F, the alloy formation is limited to a diffusion process and cladding degradation is negligible. At higher temperatures, the cladding penetration rate increases as shown in Figure E.8-1. As a design limit, the cladding attack is limited to less than 10% of the wall thickness, or 2 mils; the probability of cladding failure increases rapidly as the wastage exceeds 2 mils.

2. Local sodium boiling - To avoid local sodium boiling within the core, the peak coolant temperature in the core is limited to 1650 to 1900°F, depending upon local pressure conditions. A conservative saturation temperature for conditions in the core with the primary pumps not operating is 1750°F, and 1950°F is representative of the boiling point with the primary pumps operating at full flow.

3. Structural Integrity - The upper internal structure and other vessel components are protected from thermal creep damage by limiting the core average outlet temperature to the following ASME Code Level D time-at-temperature criteria.

Time at Temperature	Temperature Limit
< 1 hr	1400°F
> 1 hr	1300°F

Allowing for the stresses from the increased cover gas pressure, the ASME Code would permit operation for about 20 hours at 1500°F. Within this time, the vessel would accumulate a creep strain of about 0.5%. While the code would not permit operation above 1500°F, available creep data indicate vessel strains of more than 10% in 30 hours at 1600°F indicating possible rupture of the canopy seal and/or vessel. While pressure release through local failure may reduce stresses and extend the vessel creep life, even higher temperatures will certainly cause a potential failure and cover gas release.

4. Fuel Melting - Fuel melting, per se, is not a cause of pin failure. TREAT tests, especially M5 and M6, have demonstrated that extensive fuel melting (exceeding 80% of a given cross-section) does not affect the basic pin failure mechanism. Failure by eutectic penetration at the rates documented in Figure E.8-1 is the appropriate mechanistic cladding breach criterion even for pins with molten fuel in contact with the cladding.

   The peak temperatures and damage states calculated for the first four NRC events are summarized in Table E.8-1. The events are evaluated from a revised reference power level of 471 MWt, 11% greater than the original reference power of 425 MWt stated in the PSID. Each event is individually discussed in the following paragraphs.

Event 1: Inadvertent withdrawal of all control rods, without scram for 36 hours (single module) - (a) with forced cooling; (b) with RVACS cooling only

The ARIES results for Event 1a are summarized in Figures E.8-2 (core power and flow), E.8-3 (reactivity feedbacks) and E.8-4 (core temperatures). Based on detailed analysis of neutronic and reactivity calculational uncertainties, $0.36 is assumed as the total (six) rod runout worth for the n-th (commercial) PRISM reactor. This low reactivity insertion results in a peak power of 170% nominal, which reduces to an essentially steady-state value of 130% nominal within 250 sec. (The ARIES analysis assumes steam generator dryout and loss of IHTS cooling at about 480 sec; the subsequent increase in the lower grid plate temperature greatly reduces the power for the remainder of the transient.)

As shown in Figure E.8-4, this event is quite benign. Fuel-clad interface temperatures peak briefly at 1340°F and the maximum cladding wastage is less than 0.1 mils. No cladding failures are anticipated. Local coolant temperatures are less than 1300°F and the peak core average outlet temperature is less than 1250°F.

Event 1b, with only RVACS available for heat removal, i.e., less of IHTS cooling at time zero versus at 480 seconds as assumed in Event 1a, is significantly more severe. (See Figures E.8-5, E.8-6 and E.8-7.) Most fuel pins in the 12 inner ring, peak powered driver fuel assemblies will fail during the 36-hour transient but there will be no loss of structural integrity and no coolant boiling.

The power peaks at 170% nominal at 23 sec, and the primary EM pump windings overheat and short out, failing the pumps, at 120 sec, with the core average outlet temperature greater than 1300°F (Figure E.8-5). Peak temperatures then occur within 20 sec due to the sudden drop in flow and a short period of high power-to-flow ratio which follows. The maximum fuel-clad interface temperature peaks at 1650°F at 150 sec, the maximum coolant temperature peaks at 1640°F, also at 150 sec, and the core average outlet temperature peaks at 1530°F at 140 sec and drops to the range of 1280 - 1300°F for the remainder of the event (Figure E.8-7), as a zero reactivity equilibrium state is achieved at this level (Figure E.8-6).

TABLE E.8-1
PEAK TEMPERATURES AND DAMAGE EXTENT
DURING NRC DETERMINISTIC EVENTS

Event	Peak Temperatures (F) Fuel-Clad Interface	Local Coolant	Core Avg Outlet
1 - All Rods UTOP			
1a - w/ forced cooling	1340	1300	1250
1b - w/ RVACS only	1650	1640	1530
2 - Station blackout for 16 hrs (ULOF/LOHS)	1350	1350	1220
3 - Loss of forced cooling + loss of RVACS, with scram	1630 at 36 hrs	1630 at 36 hrs	1610 at 36 hrs
4 - ULOF/LOHS with instant. loss of flow, one pump	1420	1420	1235

Event	Extent of Damage Sodium Boiling	Loss of Structural Integrity	Pin Failures
1a	No	No	None, <0.1 mil clad attack
1b	No	No	Pins in 12 inner fuel assemblies will fail
2	No	No	None, <0.1 mil clad attack
3	No	No	All fail in 22-23 hrs
4	No	No	None, <0.2 mil clad attack

The short core outlet excursion above 1400°F, although violating the ASME Level D limit, does not threaten structural integrity. The sodium boiling temperature (1750°F after loss of flow) is not approached.

Many cladding failures in the 12 inner fuel assemblies are expected, however. The brief excursion to 1650°F on the peak pins is calculated to waste less than 1 mil of cladding during the first 500 sec. Peak fuel-clad interface temperatures then drop to the range of 1290 - 1305°F for the remainder of the transient. At 1300°F, the cladding attack rate is 0.16 x 10^{-3} mil/sec. Wastage of a second mil of cladding requires 1.74 hours. Ten mils of cladding (half-thickness) is lost in less than 16 hours. Therefore, cladding failures can be expected after two hours into the transient and most of the pins in the 12 inner fuel assemblies will have failed in 16 hours.

The long-term soak at 1290-1305°F with ruptured fuel pins presents an additional potential problem. The fuel-clad eutectic is formed of 5-6 times as much fuel as cladding on a volumetric basis. At the same time that the eutectic has consumed about 8 mils of cladding, <u>all</u> of the fuel at that elevation has gone into the molten eutectic. Thus, the potential exists, after pin failure, for molten fuel as a component of the eutectic to be swept out into the primary sodium. The top 9% of the fuel column is above 1290°F surface temperature and can become molten. The physical consequences to the core are not known. Further evaluation of existing experimental data, and potentially new data, will be required to develop an adequate characterization of physical behavior under these conditions. Such efforts have been initiated at Argonne National Laboratory.

Event 1b, the all-rods UTOP with RVACS cooling only, has also been investigated under the assumption the primary pumps are tripped at zero time, rather than running until they fail from overheating. This produces a somewhat more benign short-term transient as the more rapid increase in coolant and cladding temperatures turn the power spike over more quickly. However, the consequences of this variation of the event are essentially the same. Cladding failures can be expected after two hours into the transient with most of the pins in the 12 inner fuel assemblies failing in 12 hours. Boiling does not occur and structural integrity is preserved (the Level D limit is not exceeded). The following table summarizes the key differences between the two variations.

	With Pump Failure from Overheating	Pumps Tripped at Zero Time
Peak Power (% Initial)	170	130
Peak Fuel Temperature (F)	1880	1670
Peak Fuel-Clad Interface Temp (F)	1650	1550
Peak Local Coolant Temp (F)	1640	1550
Avg Core Outlet Temp (F)	1530	1360
Long Term Peak Fuel-Clad Interface Temperature (F)	1305	1340

Event 2: Station blackout for 16 hours, with loss of all ac power

The total loss of site power is assumed to result in the loss of pump flow and IHTS cooling. For the module with failure to scram, this is an unprotected loss of flow and heat sink event (ULOF/LOHS). The ARIES results are shown in Figures E.8-8, E.8-9 and E.8-10. The primary flow reduces more quickly than the power, resulting in a power-to-flow mismatch. Natural circulation is initiated in the reactor vessel, and as the power reduces, the event transitions into essentially a RVACS decay heat removal transient.

As shown in Figure E.8-10, there is a near-term (100 sec) peak in fuel and clad temperatures and a long-term (>20 hr) coolant temperature peak. Peak fuel-clad temperatures do not exceed 1350°F and are above 1290°F for less than 200 sec; maximum clad wastage is less than 0.1 mils. Long-term peak cladding and coolant temperatures are less than 1250°F. The event is benign, with no predicted fuel pin failures, no coolant boiling and no structural damage.

Event 3: Loss of forced cooling plus RVACS with scram (single module), followed by 25% partial unblockage of RVACS after 36 hours

This extremely low probability transient assumes the essentially impossible event of the combined loss of all three heat removal systems: the normal steam generator-water train, the Auxiliary Cooling System (ACS) on the steam generator and the complete loss (total inlet and outlet blockage) of the fully passive, safety grade RVACS. It should be noted that complete blockage of all RVACS inlet and outlet passages is essentially impossible; physical arguments supporting this assertion are given in Addendum A. As pointed out in Addendum A, a blockage at the bottom of the RVACS by, say, sand or mud up to above the bottom of the collection cylinder does not result in the total blockage of the RVACS. Rather, a natural circulation pattern will be set up in the normally upflow annulus between the containment vessel and the collecting cylinder. The maximum average core outlet temperature will be only 1155°F, 47°F above the maximum temperature for the nominal RVACS transient. The only way the RVACS can be completely lost is to physically block all inlets and all outlets; it is anticipated that such a blockage could be readily removed in 8 hours rather than requiring 36 hours as specified by the event definition.

Event 3 has been analyzed in two steps. The first 2.5 hours (9000 sec) have been calculated by the ARIES plant transient analysis code assuming (1) the scram inserts $16 of negative reactivity, and (2) there are no heat losses at all (i.e., no losses into the silo surrounding the reactor vessel). The resulting power and flow time histories are shown in Figure E.8-11 and core temperatures in Figure E.8-12. The scram upon loss of cooling is seen to result in low temperatures at the beginning of the transient (i.e., <800 F for the first hour). However, without any decay heat removal systems, the core decay power is not balanced by heat removal and the bulk temperature of the reactor slowly rises.

The transient was continued out to 36 hours by means of a second model especially created to analyze this event. The model is a thermal R-C nodal network which accounts for:

- Radiation from the reactor vessel to the containment vessel
- Radiation from the containment vessel to the collector cylinder
- Radiation from the collector cylinder to the silo wall
- Conduction outward through the silo wall and surrounding earth

Although of minor importance, the heat rejection from the bottom of the reactor vessel has also been included. Heat losses through the top closure and from the IHTS piping are neglected.

The results of this analysis are shown as peak and average reactor vessel temperatures in Figure E.8-13. The peak reactor vessel temperature is identical to the mixed mean core outlet temperature. At 36 hours, the mixed mean core outlet is 1610°F. With the decay heat reduced to 0.5% of full power (2.33 MWt) and with a correspondingly low natural circulation flow through the core, the radial temperature peaking is nominal. The peak local sodium temperature is less than 1630°F and boiling does not occur.

As cladding temperatures increase above 1290°F during the transient, eutectic attack of the cladding will become more and more rapid. Cladding failures, of the high burnup pins, are anticipated to begin by 1350°F (17 hours into the transient) by cladding strain due to fission gas loading, and essentially all driver fuel and blanket pins will have failed by 1450°F (23 hours into the transient). The last pins to fail will be low burnup pins without significant fission gas generation. The failures will be by cladding strain due to internal pressure after extensive (10-15 mils) cladding wastage by fuel-clad eutectic.

The structural consequences of the temperature increases from the loss of forced cooling and RVACS blockage are the increased creep rates and decreased creep rupture times. These effects are accentuated by the increase in the cover gas pressure with the fission gas release from the failed fuel pins. The failure modes promoted by these elevated temperatures and pressures are: 1) creep relaxation of the closure plate and plug attachment bolts which would pressurize and may rupture the seal-weld canopies, and 2) creep rupture of the reactor vessel.

Assuming the transient event takes place shortly before refueling (i.e., at maximum burnup on 1/3 of the fuel) the total volume of fission gas released to the cover gas plenum is 15.5×10^6 cc at STP conditions. This is sufficient, when combined with the reactor sodium expansion at this temperature, to repressurize the plenum to approximately 90 psig.

The reactor head temperatures will remain below ~1050°F throughout the event permitting direct application of the ASME Code which indicates that the bolt pre-loads will be maintained precluding pressurization and rupture of the canopy seals. The reactor vessel temperatures will exceed the ASME Code limit of 1500°F for about 16 hours requiring extrapolation of the rupture times and isochronous curves given in the code. Such extrapolation indicates a creep damage of ~0.075 for the event compared to allowable creep fatigue damage of 1.0. Even after allowing for a ~0.1 damage accumulation from the PRISM normal duty cycle, the predicted creep damage is sufficiently small to envelop any uncertainties in the extrapolation.

Thus, the loss of RVACS for 36 hours will not rupture the cover gas pressure boundary. The reactor vessel will accumulate substantial strains. However, with the control rod scrammed, the deflections associated with the strains will not have any safety impact.

The long-term soak at temperatures above 1300°F (for 21 hours plus cooldown time) with ruptured fuel pins presents the additional potential problem of molten fuel, as part of the eutectic, being released into the primary sodium. From about 24 hours on to the end of the transient, the core inlet sodium temperature is greater than 1300°F and the fuel-clad eutectic will remain liquid. The physical consequences to the core of remaining in this condition where the whole core is above the fuel-clad eutectic formation point are not know. Further evaluation of existing experimental data, and potentially the attaining of new data, will be required for an adequate characterization of physical behavior under these conditions. Evaluations of this problem are underway at Argonne National Laboratory.

<u>Event 4</u>: Instantaneous loss of flow from one primary pump; coastdown flow for other pumps; without scram for 36 hours

This has been interpreted as a 36-hour station blackout, resulting in loss of flow and loss of IHTS heat sink, with the addition of loss of the synchronous machine connection on one primary EM pump resulting in instantaneous stoppage of flow (no coastdown) for this pump. As such, this event is a more severe version of Event 2.

The ARIES results are summarized in Figures E.8-14, E.8-15 and E.8-16. The effect of the instantaneous flow stoppage on one pump is seen only in the first 500 sec of the transient while the other three primary pumps coast down. The short-term (<100 sec) peak cladding and coolant temperatures, which were about 1350°F for Event 2, are increased to about 1420°F. The long-term natural circulation behavior is unaffected.

The event is benign. Peak cladding wastage by eutectic attack is less than 0.2 mils; no pin failures are expected. Margins to local coolant boiling and structural damage are essentially the same as for Event 2. The core average outlet temperature does not exceed 1250°F.

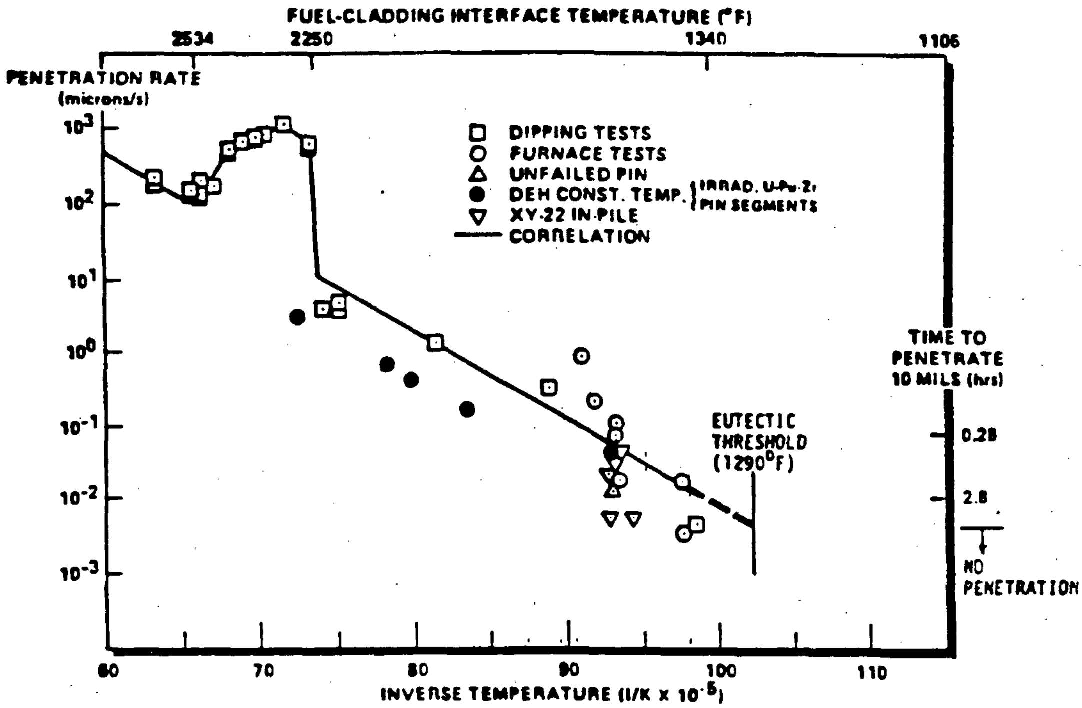


Figure E.8-1 - Rate of Cladding Penetration by Metal Fuels

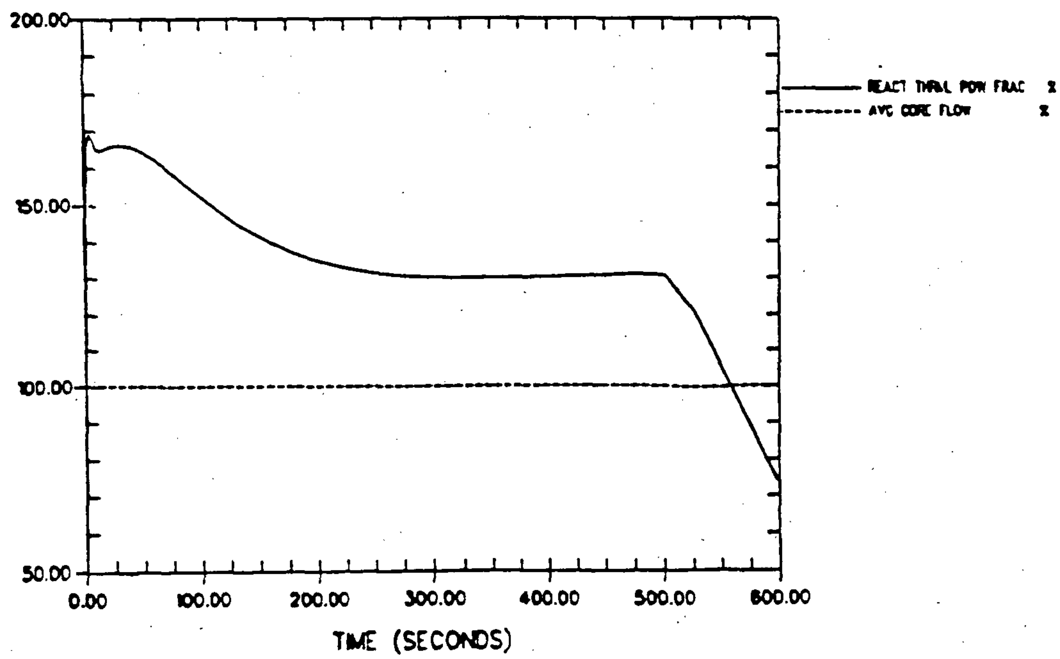


Figure E.8-2 - Event 1a, All Rods UTOP with Forced Cooling: Core Power and Flow

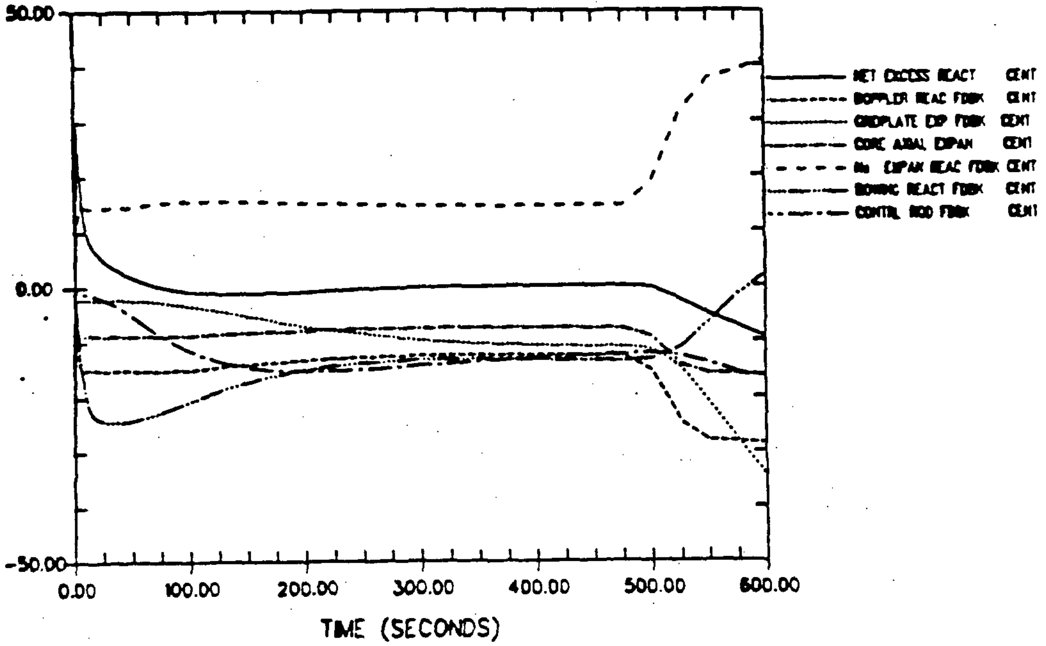


Figure E.8-3 - Event 1a, All Rods UTOP with Forced Cooling: Reactivity Feedbacks

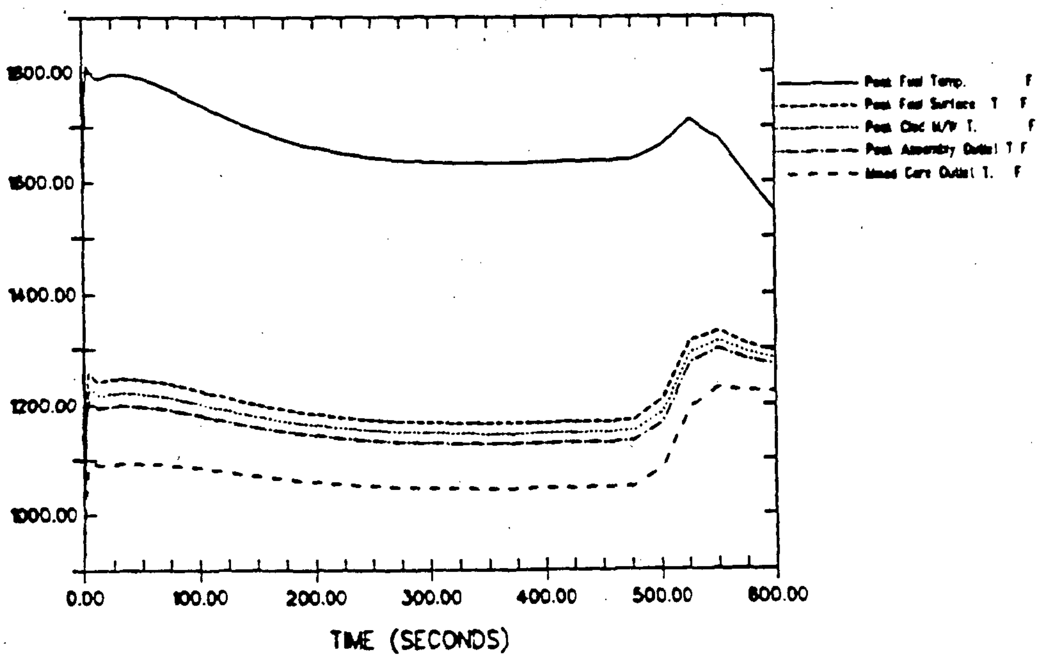


Figure E.8-4 - Event 1a, All Rods UTOP with Forced Cooling: Core Temperatures

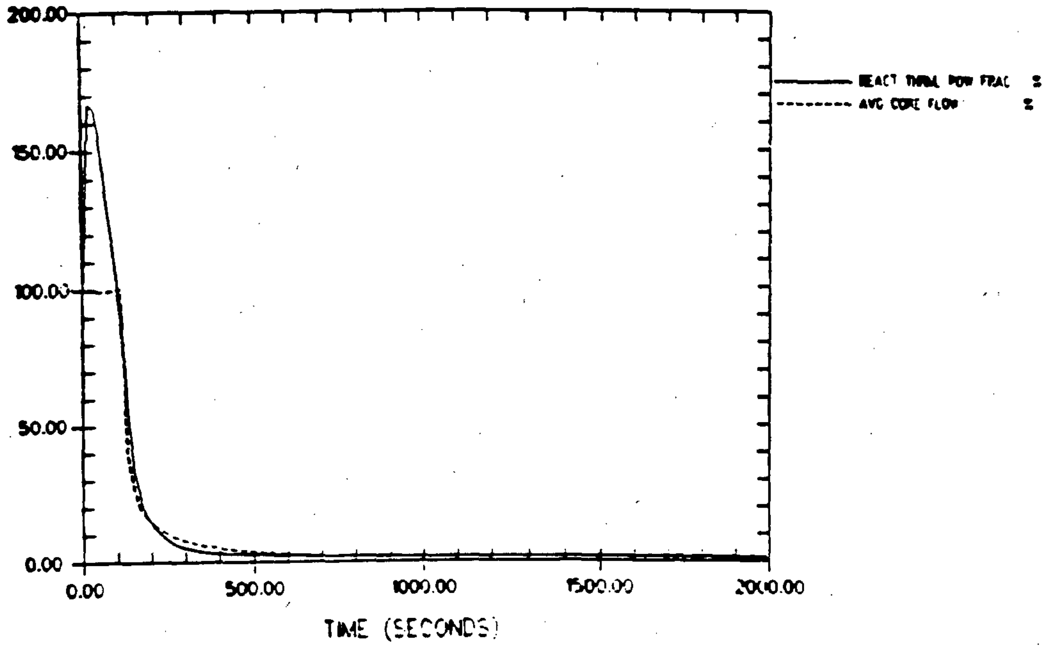


Figure E.8-5a - Event 1b, All Rods UTOP with RVACS Cooling: Core Power and Flow

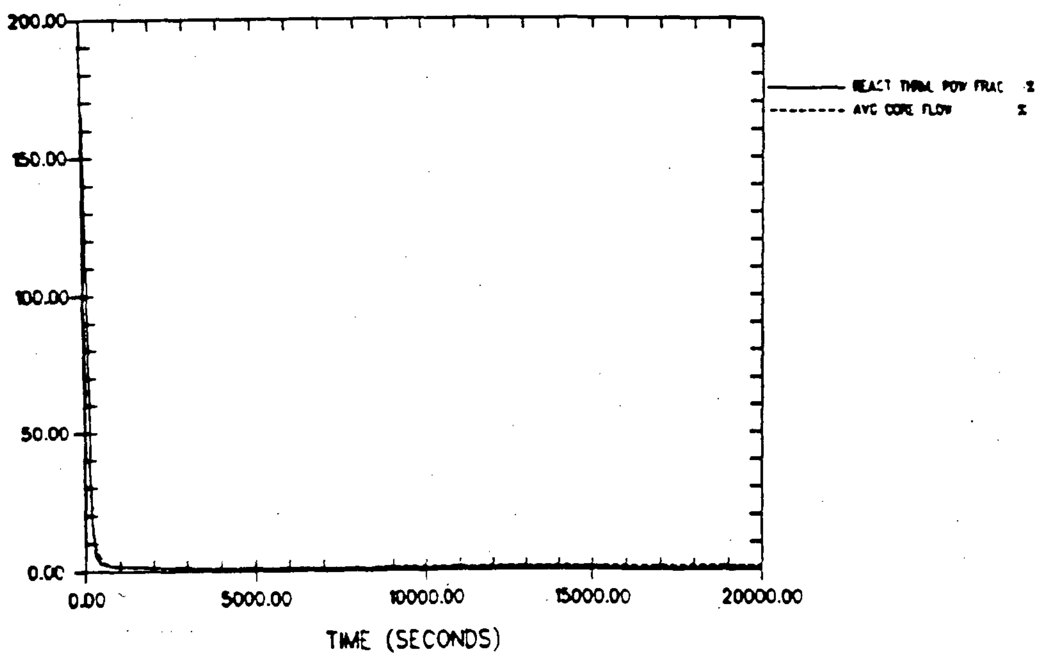


Figure E.8-5b - Event 1b, All Rods UTOP with RVACS Cooling: Core Power and Flow

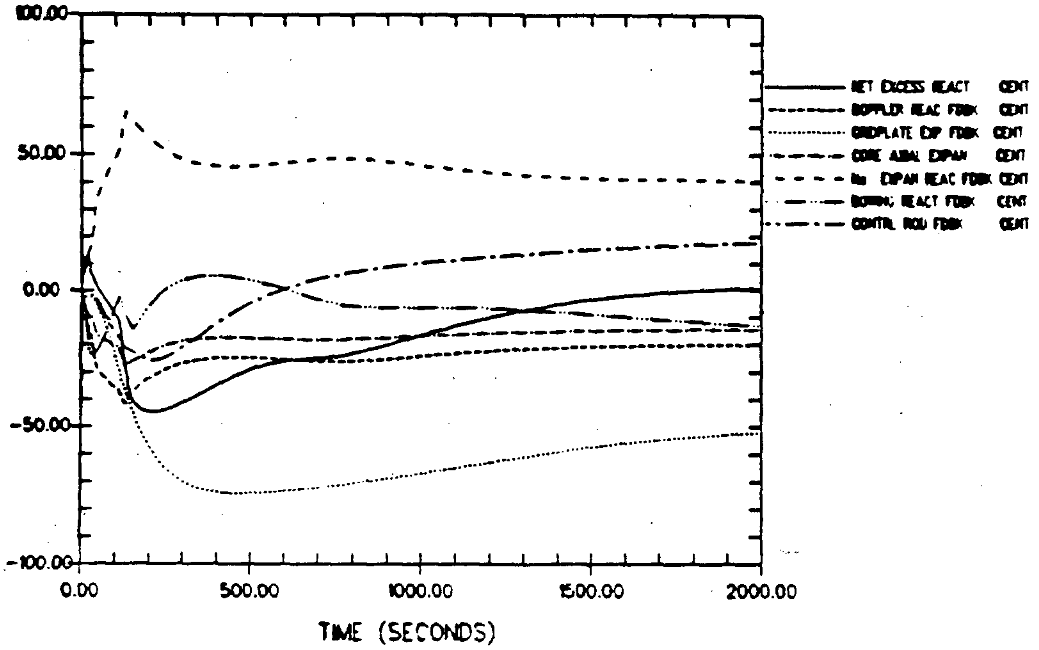


Figure E.8-6a - Event 1b, All Rods UTOP with RVACS Cooling: Reactivity and Feedbacks

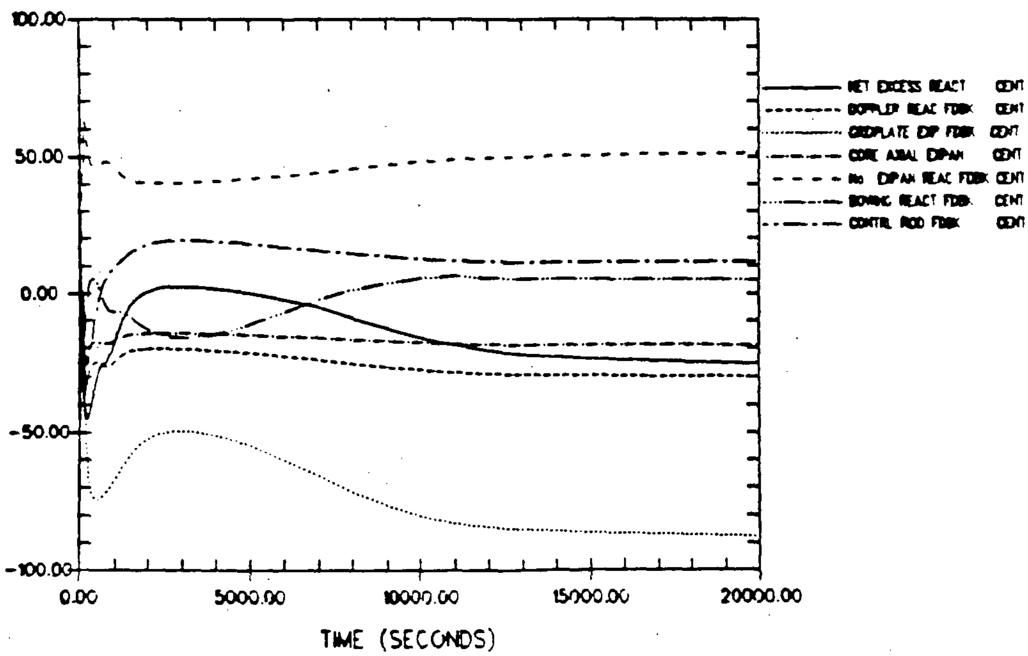


Figure E.8-6b - Event 1b, All Rods UTOP with RVACS Cooling: Reactivity and Feedbacks

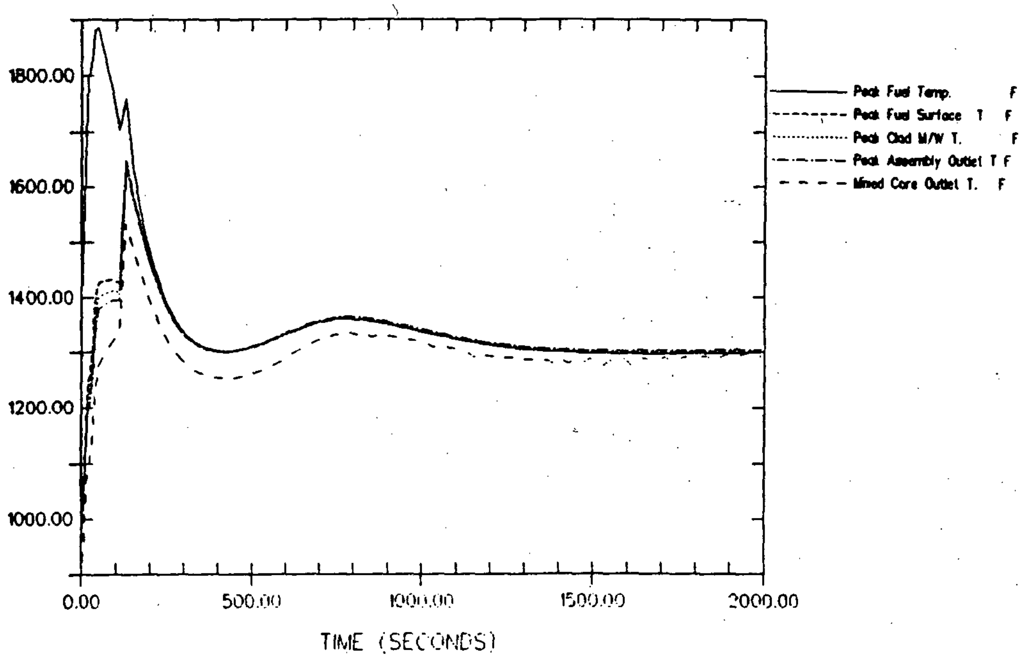


Figure E.8-7a - Event 1b, All Rods UTOP with RVACS Cooling: Core Temperature

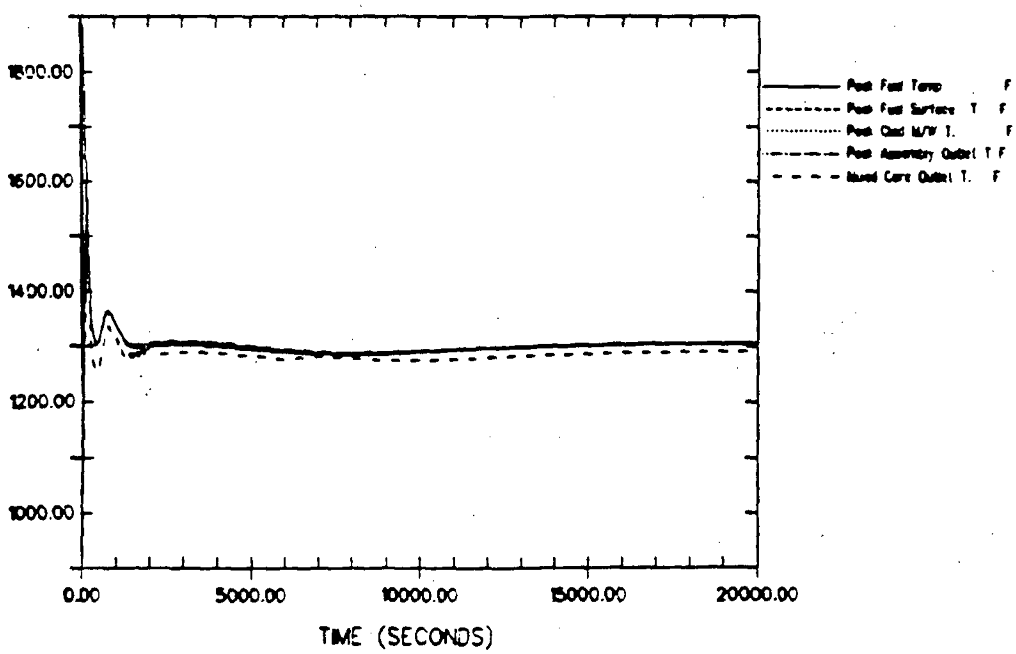


Figure E.8-7b - Event 1b, All Rods UTOP with RVACS Cooling: Core Temperature

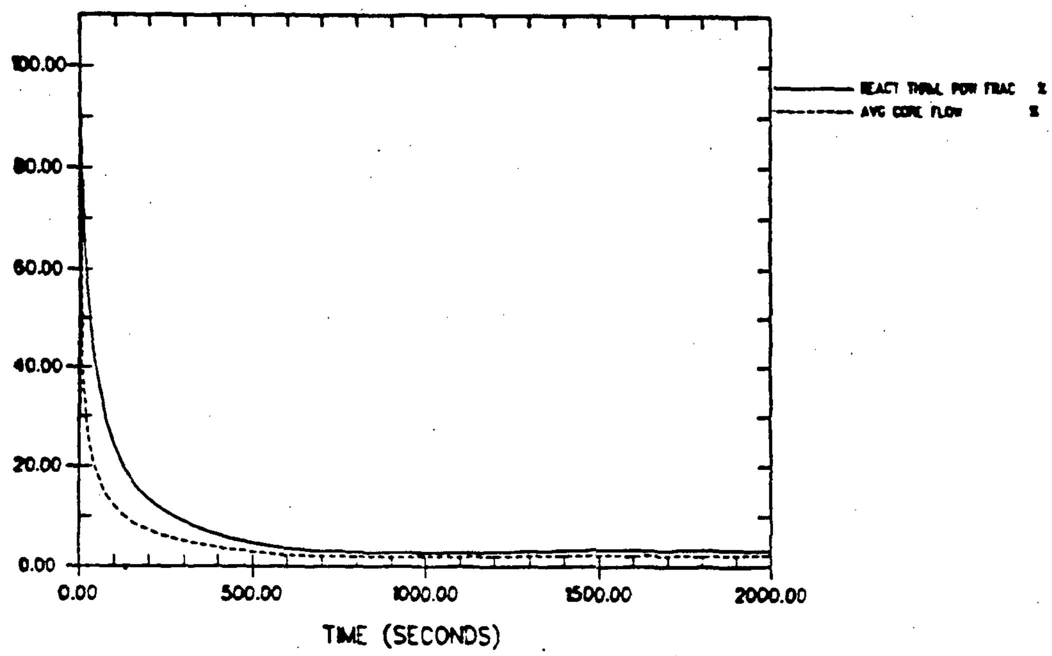


Figure E.8-8 - Event 2, Station Blackout: Core Power and Flow

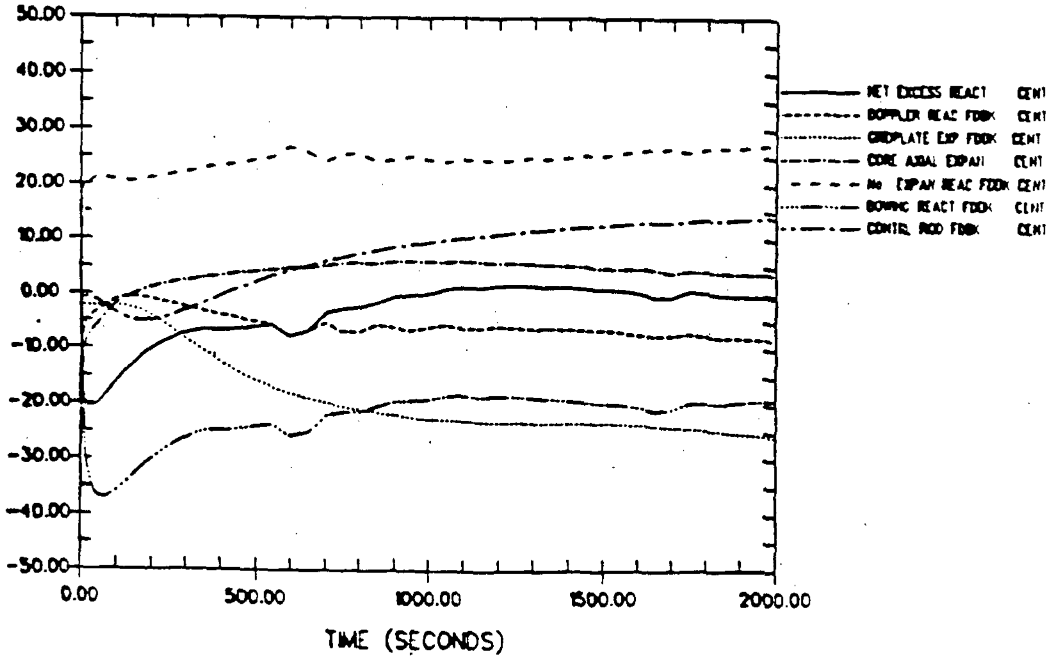


Figure E.8-9 - Event 2, Station Blackout: Reactivity Feedbacks

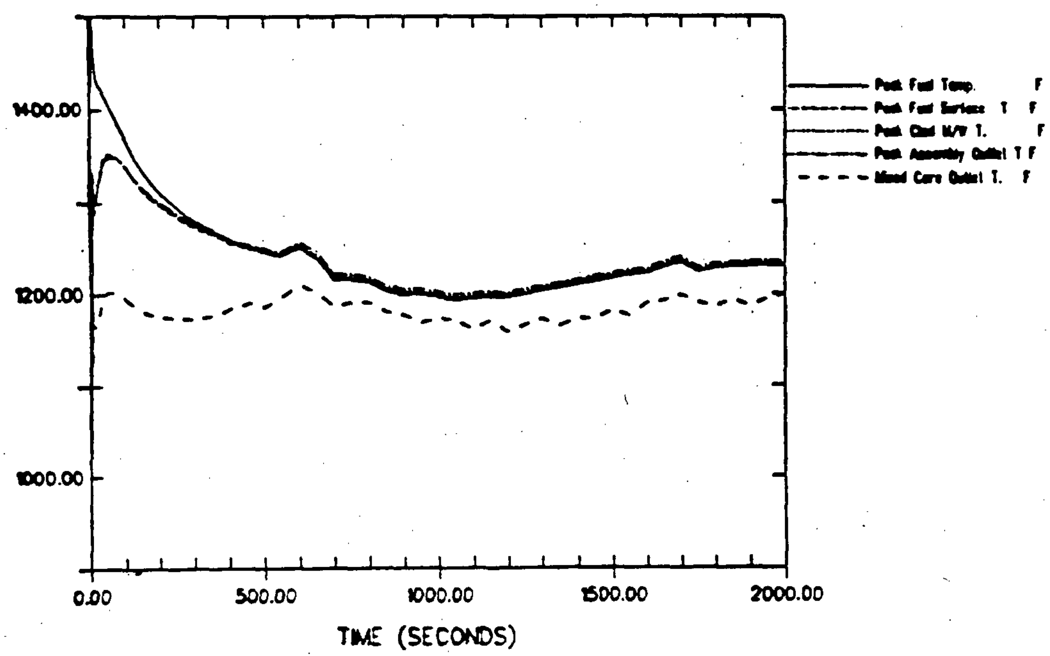


Figure E.8-10 - Event 2, Station Blackout: Core Temperature

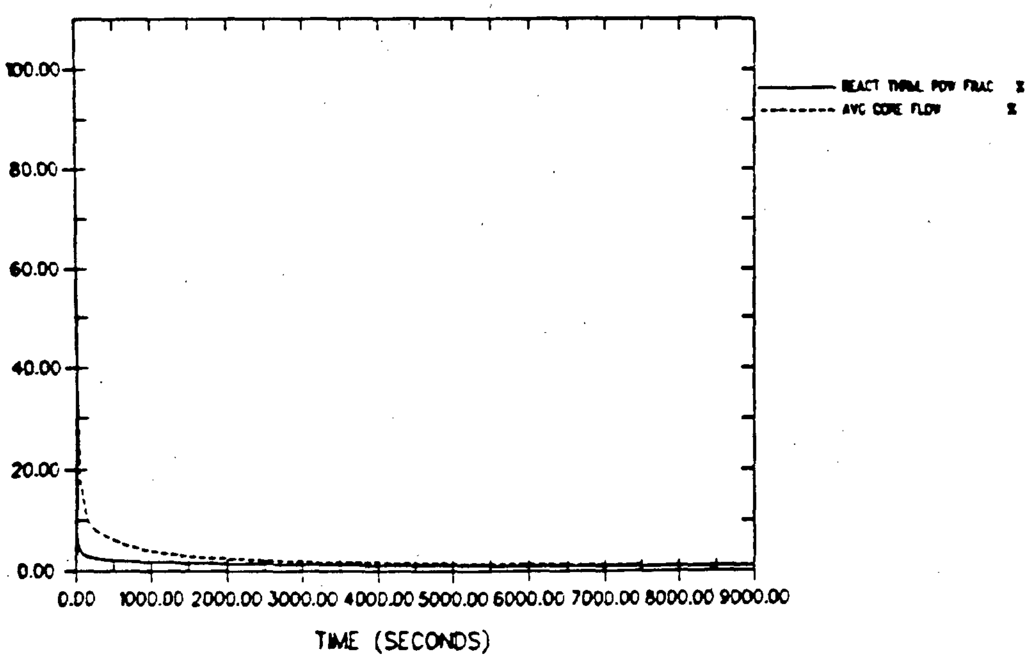


Figure E.8-11 - Event 3, Loss of All Cooling: Core Power and Flow

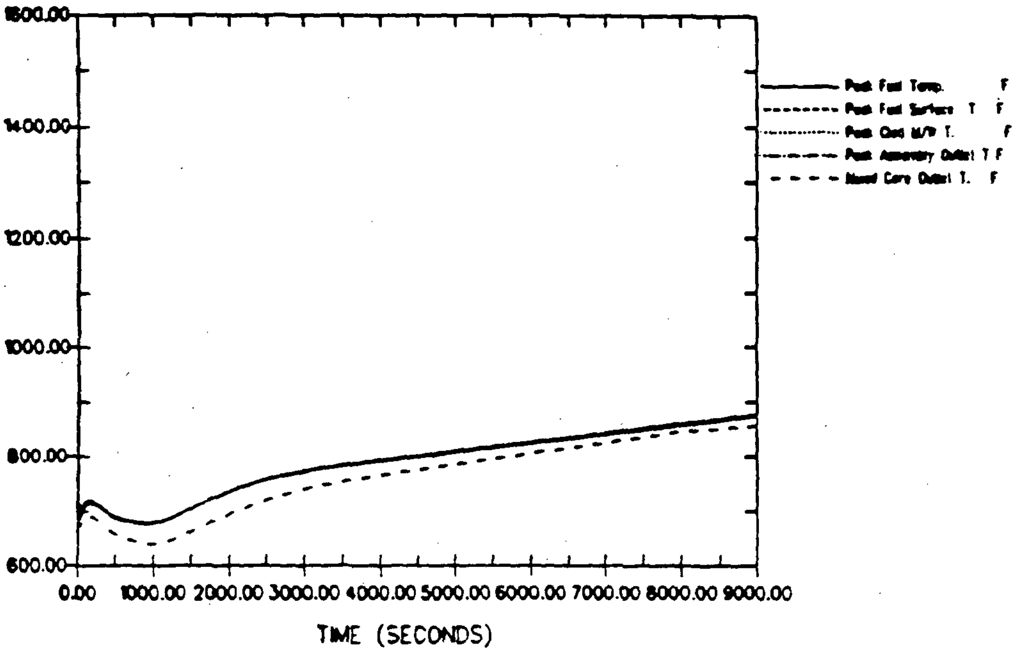


Figure E.8-12 - Event 3, Loss of All Cooling: Core Temperatures

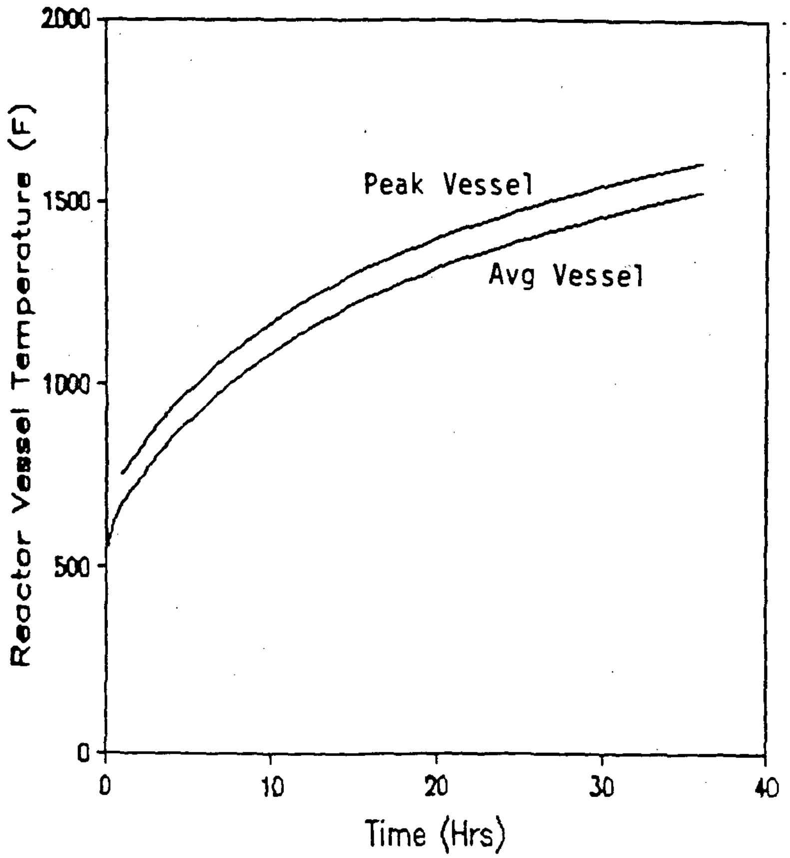


Figure E.8-13 - Event 3, Loss of All Cooling: Long-term Core Temperatures

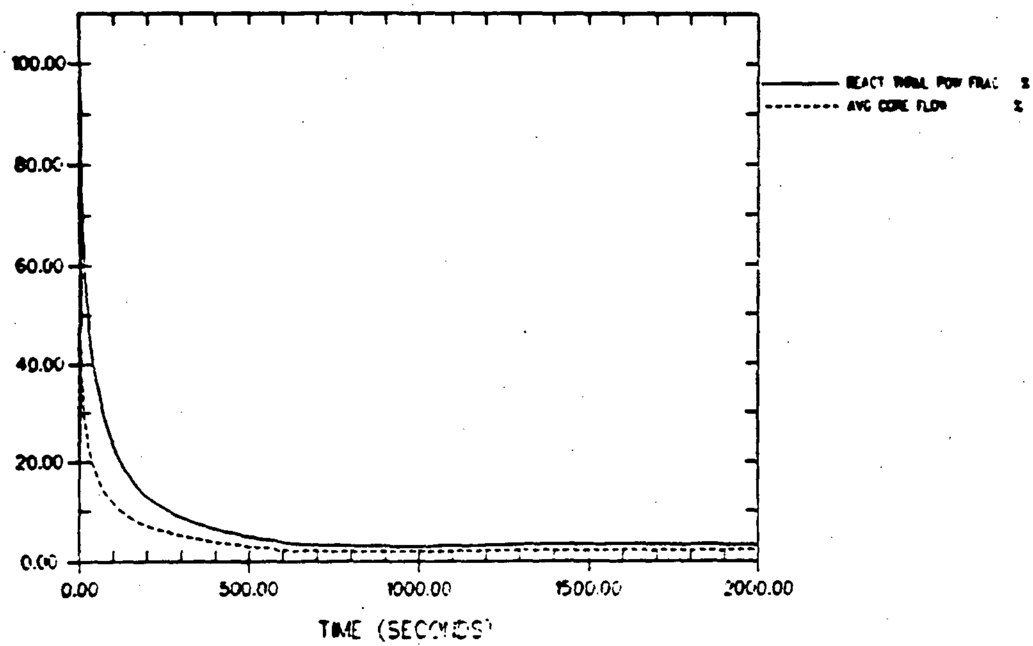


Figure E.8-14 - Event 4, ULOF/LOHS with 3 Pump Coastdown: Core Power and Flow

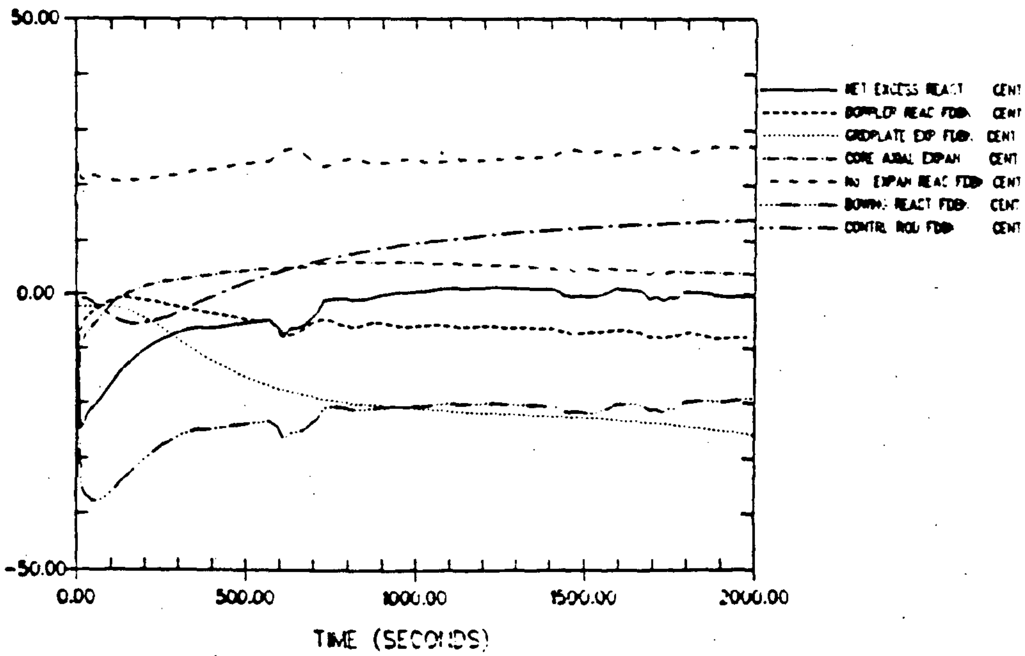


Figure E.8-15 - Event 4, ULOF/LOHS with 3 Pump Coastdown: Reactivity Feedbacks

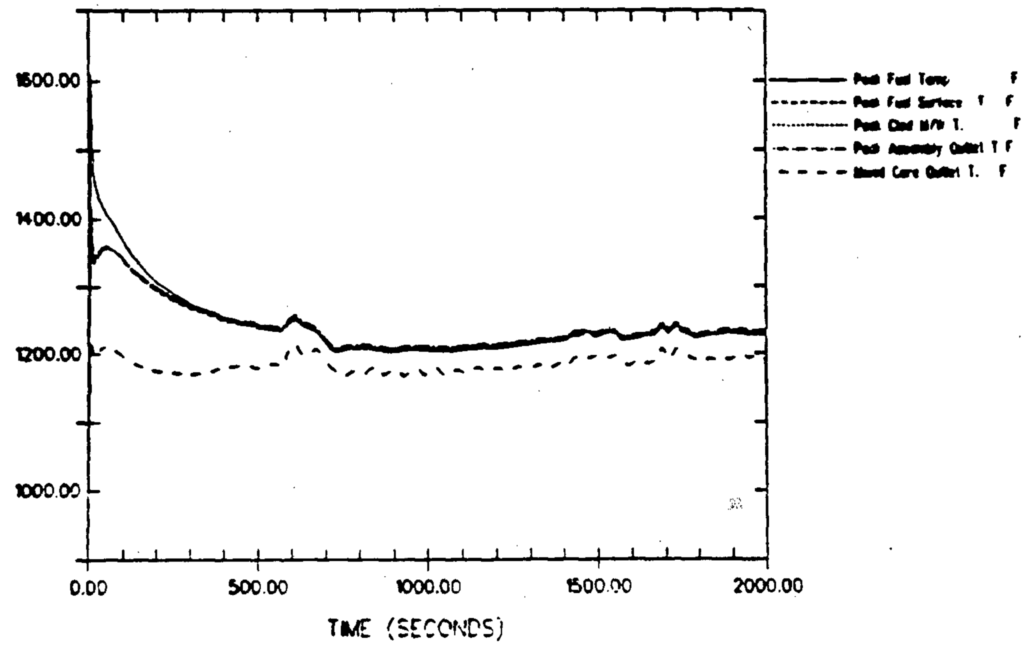


Figure E.8-16 - Event 4, ULOF/LOHS with 3 Pump Coastdown: Core Temperatures

Event 5 Steam generator tube rupture with failure to isolate or dump the water from the steam generator:
- **Justifiable number of tube failures**
- **Defined sequence of ruptures**

The PRISM steam generator system includes redundant, quick-acting steam/water isolation and blowdown valves at each steam generator for rapid isolation and complete dump of the water within 60 seconds. These valves may be actuated by the operator, and they are triggered automatically by redundant control circuitry upon bursting of the 28-inch diameter sodium-side rupture disc located at the bottom of each steam generator. The dump of IHTS sodium by the rupture disc bursting will result in the reactor protection system scramming the reactor if the operator has not already initiated a scram. In order to control and limit sodium-water reaction damage during a steam generator tube failure event, it is essential that the steam/water isolation and blowdown system function. Therefore, this system will be designed for a very low failure probability, in the order of 10^{-5} failures per demand.

Uncontrolled release of steam/water into the sodium from failure of the steam/water isolation and blowdown system after a tube leak will result in progressive failure of more steam generator tubes, spread of sodium/water reaction damage, and rise in shell-side pressure from the volume of hydrogen generated and the unreacted steam in the shell. The quasi-steady state peak pressure that would be reached if all the tubes in the steam generator eventually burst is estimated to be about 860 psi for the PRISM design. The IHX is designed to withstand the full 1000 psi system steam pressure for at least one hour without exceeding the ASME Code, Section III, Level D strain damage limits. Thus, no breach of the IHX from overpressure conditions is expected to occur, provided it is not damaged by caustic attack or sodium/water reaction in the secondary sodium side of the unit.

There is a significant initial buffer comprised of over 165 feet of sodium-filled main piping between any sodium/water reaction zone in the steam generator and the IHX which at the beginning of the event will prevent caustic reaction products and sodium/water reaction zones from reaching the IHX. However, if the steam/water injection continues uncontrolled and sufficiently high pressure differences between the hot leg and cold leg piping of the IHTS at the steam generator are sustained (about 12 psi required), then there could be a flow of sodium and an advance of the sodium/water interface from the steam generator to the IHX, in which case caustic attack, sodium/water reaction damage, and penetration to the primary side within the IHX could occur.

Termination of this extreme version of the event will be accomplished when the flow of steam/water to the IHX is halted, either by finally actuating the necessary valves to disconnect from the source in the steam/water system, by depletion of the steam/water source, or by a burn-through of the IHTS piping from the sodium/water reaction in the pipe such that the steam is vented sufficiently to stop further advance toward the IHX. Ultimately, termination activities will include inerting and capping the IHTS piping to seal the secondary side of the IHXs.

With the high reliabilities included in the PRISM steam generator system design objectives, the estimated probability that this sequence of events would happen such that the IHX would be penetrated into the primary side and a significant radioactivity release would occur is extremely low, in the order of 10^{-11} per year for a three-module power block. Thus, the event is believed sufficiently improbable that it should be classified in the residual risk category.

Probability of Event 5

The contributing probability factors required to produce this event in a manner which results in a radiological release are summarized below.

(1)	Steam generator tube leak (per unit)	10^{-3}/yr
(2)	Failure of the operator to notice the leak initiation signals coming from the redundant hydrogen sensors at the exit of the SG and the small vent line at the top of the SG unit and failure to take the necessary action to isolate and blow down the unit before the SG leak progresses sufficiently to burst the rupture disk and cause the automatic isolation and blowdown system to take action.	10^{-1}/event
(3)	Failure of the automatic isolation and blowdown system. This system includes redundant isolation valves and a check valve on the steam line, dual isolation valves and a feedwater control valve on the feedwater supply lines, and dual SG water dump lines and valves as well as power operated pressure relief valves. All these valves are automatically actuated by redundant instrumentation which detects the failure of a SG rupture disk.	3×10^{-5}/demand
(4)	Probability that a sufficient pressure differential (>12 psi) will be produced by the vented steam within the SG shell to force steam back into the IHX given the failure to terminate the event. A minimum of about 30 average size blowout tube ruptures, or their equivalent, are estimated to be required, located near the top of the steam generator for the vented steam to cause a pressure differential of about 12 psi on its way within the SG shell to the relief system.	10^{-1}/event
(5)	Failure of the operator to trip the turbine and isolate the other two steam generators in the power block. This is an additional method of shutting off the source of steam and feedwater originating from the unaffected units.	10^{-2}/event
(6)	Event multiplier for three-module power block	3
	Total probability that the event will occur and result in a significant radioactivity release, per 3-module power block	~10^{-11}/year

Analysis of Event 5

Sodium/water reaction tests conducted at ETEC with a large scale model of the CRBRP steam generator showed that if not terminated, a relatively small initiating tube leak can result in a rapidly expanding series of tube failures involving a large number of tubes.* This type of event where massive tube failures have resulted has also been experienced in operating LMR power plants (for example, BN350 in the USSR and within the last year, PFR in the UK). In each instance at the operating power plants the event was finally brought under control and terminated by shutting off the steam/water supply in parallel with relief of sodium-side pressure by rupture disc bursting and/or rapid sodium dump. In all of these plant cases many steam generator tubes ruptured before the event was brought under control. No reporting of IHX damage is known for the BN 350 nor, at this date, for PFR.

A preliminary analysis has been done of the sequence of tube failures that might occur in the PRISM steam generators after an initiating tube leak if steam/water isolation and blowdown failed to be accomplished. Early in the process the sodium pressure rises sufficiently (about 300 psi) to burst the 28-inch diameter rupture disc at the bottom of the steam generator, thereby allowing sodium, hydrogen, and unreacted steam/water to vent from the shell side. A progressively larger number of tubes are assumed to experience blowout ruptures due to overheating caused by local high sodium/water reaction temperatures (~2000°F), based on the US sodium/water reaction tests done for CRBRP.* The back-pressure caused by the vented fluids escaping through the sodium/water reaction relief system will increase the steam pressure in the steam generator shell toward full steam pressure (1000 psi) as additional tubes fail and their steam/water flow is added.

The peak back pressure depends on the venting capacity of the relief system and the degree of continued supply of steam/water from the feedwater and steam systems. The calculations assume that the automatic steam/water isolation and blowdown system completely fails, the feedwater and recirculation pumps continue to operate, and that the steam isolation valves and the check valve in the line to the main steam header fail to close, thereby allowing steam from the other two steam generators in the power block to flow into the failed steam generator. In the extreme, if <u>all</u> the tubes in the affected steam generator eventually fail, the steam pressure within the steam generator shell will increase to a maximum pressure estimated to be about 860 psi for the PRISM system. It is estimated this process would take in the order of ten minutes or more. The resulting peak pressure in the IHX (about 860 psi) is substantially below the 1000 psi ASME Level D design pressure, so a pressure induced failure of the IHX is not expected to occur.

The sodium in the steam generator, pump, expansion tank, and the main sodium pipe from the steam generator to the pump inlet will dump into the sodium/water reaction pressure relief system after rupture of the 28-inch diameter rupture disc at the bottom of the steam generator (Figure E.8-17).

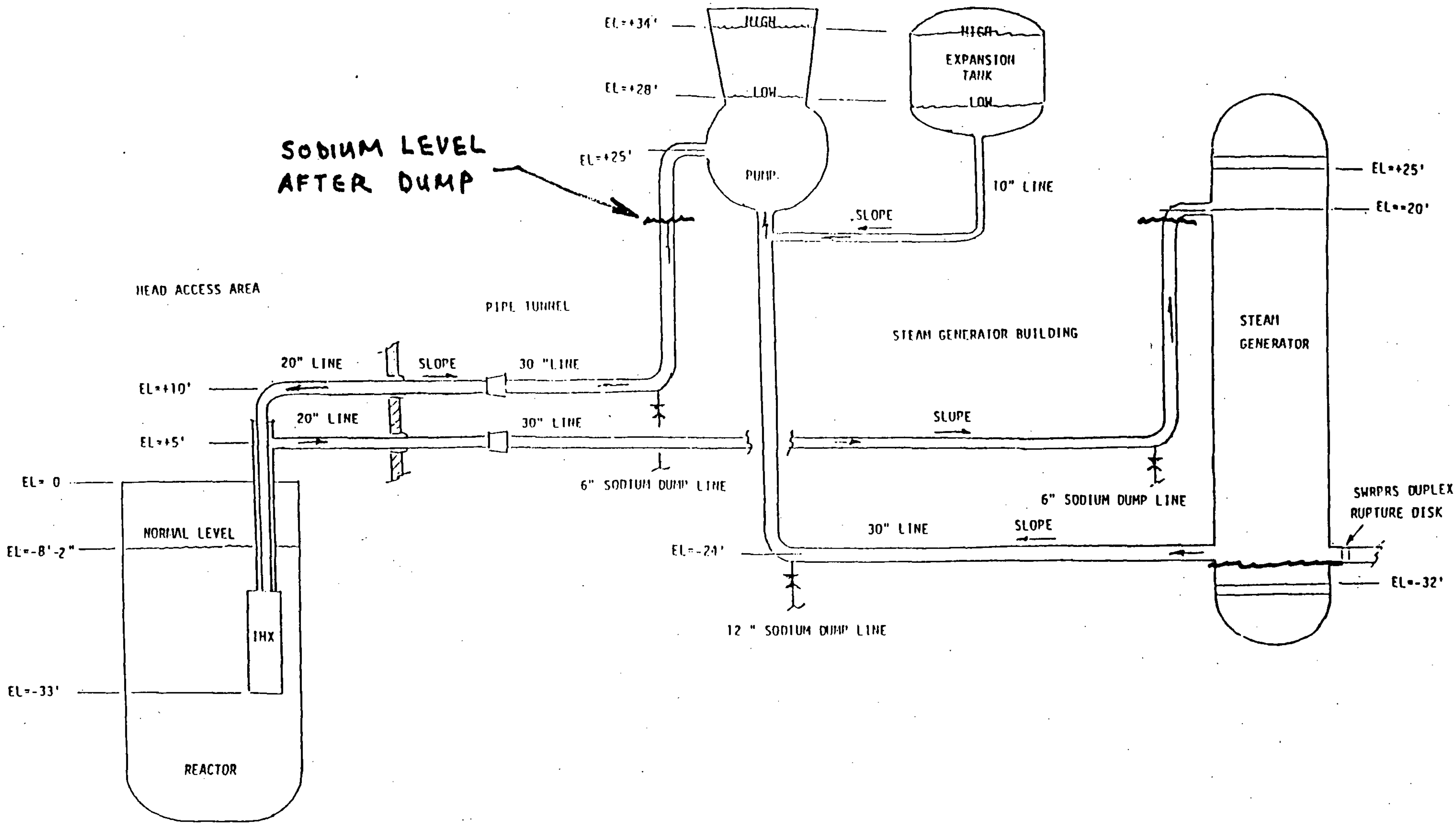
* GEFR-0063, "LLTR Series II Test Program Intermediate Leak Tests - Final Report," JC Amos and PM Magee, September 1983.

The rupture disc will rupture when the shell-side pressure reaches 300 psi. There is a total of about 50,000 gallons of sodium in the IHTS and all of it will drain through the rupture disc nozzle except that trapped by the vertical sections near the pump discharge and the steam generator inlet, respectively, and the connecting horizontal sections back to the IHX (about 14,000 gallons, maximum). Both of the vertical lines are more than 100 feet from the IHX risers and downcomers, and due to the IHTS expansion loops the length of sodium-containing piping that separates the IHX from the vertical sections next to the steam generator is more than 165 feet. This separation provides a large initial protective buffer between the IHX and any sodium/water reactions in the steam generator.

The vertical sections of the IHTS will result in the sodium level stabilizing after the dump at the elevations shown in Figure E.8-17, provided there is no pressure difference imposed between the two main sodium pipe legs. The points labeled "sodium level" in Figure E.8-17 are interfaces between the liquid sodium remaining in the IHTS and the gases and vapors, including unreacted steam/water, in the steam generator and connecting piping. Any pressure difference imposed between the two main IHTS piping legs at the steam generator will result in a movement of the sodium and the sodium interface toward the IHX.

For example, continued injection of steam/water from burst tubes near the top of the steam generator will result in the pressure at the sodium inlet nozzle (where a static sodium level is shown in Figure E.8-17) being higher than at the discharge nozzle at the bottom of the steam generator by an amount approximating the pressure drop for the leaking steam/water and any hydrogen from the associated sodium/water reaction to flow through the shell side of the tube bundle to the rupture disc nozzle at the bottom of the unit. This pressure difference will cause the residual sodium in the pipe to flow back to the IHXs, moving the sodium interface in that direction. The sustained pressure difference required to move the sodium interface through the piping to locations inside the IHX tube bundles and thereby make it possible for steam/water to reach those locations is estimated to be about 12 psi. It is estimated that in the order of about 30 typical tube ruptures near the top of the unit, due to the high local temperatures (~2000°F) experienced during sustained large sodium/water reactions, could result in a pressure difference of about this magnitude.

Continued presence of steam/water in the IHX tube bundle could cause sufficient damage there to penetrate into the primary side. Further evaluation is required to assess the probability of this happening, beyond the very preliminary results given in the preceding section, and to assess the potential radiological consequences. The very low probability of significant radioactive release currently estimated for the event ($\sim 10^{-11}$ per year for a three-module block) suggests that the event ought to be in the residual risk category and that evaluations of it be in conjunction with the probabilistic risk assessment.



IHTS HYDRAULIC PROFILE

Figure E.8-17

Event 6a Double ended guillotine (DEG) break of the IHTS Pipe

The intermediate heat transport system (IHTS) pipe is heavy wall (~1 inch thickness) piping fabricated from ductile 304 and 316 stainless steel material and normally operates in a pressure range of 30-75 psig. The piping is fabricated from seamless stock with full penetration butt welds which are visually, surface and radiographically inspected and helium leak tested. Because of the low stress, low energy nature of the IHTS, a leak-before-break type of failure is expected for the piping. A double ended guillotine break of the IHTS is a very remote possibility and the probability for such an event is estimated at 10^{-8} occurrences per module year for the piping and 10^{-6} occurrences per module year for the piping bellows expansion joints.

Detection of sodium fires is accomplished by smoke, aerosol, and/or leak detectors. These detectors actuate alarms to alert the plant operators of the existence and location of a fire. The plant will be shut down in response to a continued indication of a sodium leak. Large sodium leaks would result in automatic plant shutdown due to low sodium level in the IHTS. The IHTS is equipped with three sodium dump lines which can drain the system in about ten minutes to reduce sodium spillage in the event of a small leak.

In the reactor head access area (HAA) the IHTS piping is enclosed in a carbon steel guard pipe to prevent a sodium spill into the HAA in the event of a pipe rupture. In the IHTS pipe tunnel any sodium spill is collected in a catch pan and drained by gravity through a vertical 12 inch diameter pipe to the steam generator building bottom catch pan. The steam generator building is equipped with a catch pan and fire suppression deck. The fire suppression system provides a means for collecting the spilled sodium to prevent chemical reactions with concrete, to suppress pool burning and limit the amount of sodium aerosols generated. This structure consists of an insulated steel container (the catch pan) with a corrugated steel cover with drain and vent pipes (the fire suppression deck). During a spill, the sodium pours onto the fire suppression deck and flows through the drain pipes into the catch pan. The bulk of liquid sodium spill is thereby isolated from the cell atmosphere. The pool burning area is limited to the relatively small surface area of the drain pipes. This system will prevent contained pool burning and limit the burning to about 2% of the spill. A double ended guillotine break of the 30 inch diameter IHTS piping would result in a large spill of secondary sodium into the IHTS pipe tunnel or into the steam generator building. The maximum amount of IHTS sodium drainable from the system during a pipe break accident is 44,000 gal. The amount of sodium spillage from the IHTS depends on the location of the pipe break. There are three low elevations in the IHTS piping each of which would result in a different sodium spill volume. Pipe breaks at each of these elevations is discussed below.

a) Pipe Break - Steam Generator Outlet Line

A major pipe break in the 30 inch diameter steam generator sodium outlet line would result in the largest sodium spill of ~44,000 gal. The pipe break location and sodium levels in the system after drainage are shown on the IHTS Hydraulic Profile drawing in Figure E.8-18. The break occurs at elevation -24 feet and drains the IHTS pump, expansion tank and steam generator. The intermediate sodium in the IHX inlet and outlet lines below elevation +20 feet will not be drained. About 10,000 gal. of sodium will remain in these lines and the two IHX units. The sodium-air interface after the break will be in the steam generator building piping which is over 150 feet from the IHX.

b) Pipe Break - Pump Discharge Line

A major pipe break in the 30 inch diameter IHTS pump discharge line would result in a sodium spill of ~19,500 gallons. The pipe break location and sodium levels in the system after drainage are shown on the IHTS Hydraulic Profile drawing in Figure E.8-19. The break occurs at elevation +10 feet and drains the IHTS pump, expansion tank and upper portion of the steam generator. The intermediate sodium in the steam generator and steam generator inlet line below +20 feet elevation and in the IHX and IHX piping below +10 feet elevation will not be drained. The sodium-air interface in the IHX inlet line is in the HAA directly above the IHX inlet nozzle after the break.

c) Pipe Break - Steam Generator Inlet Line

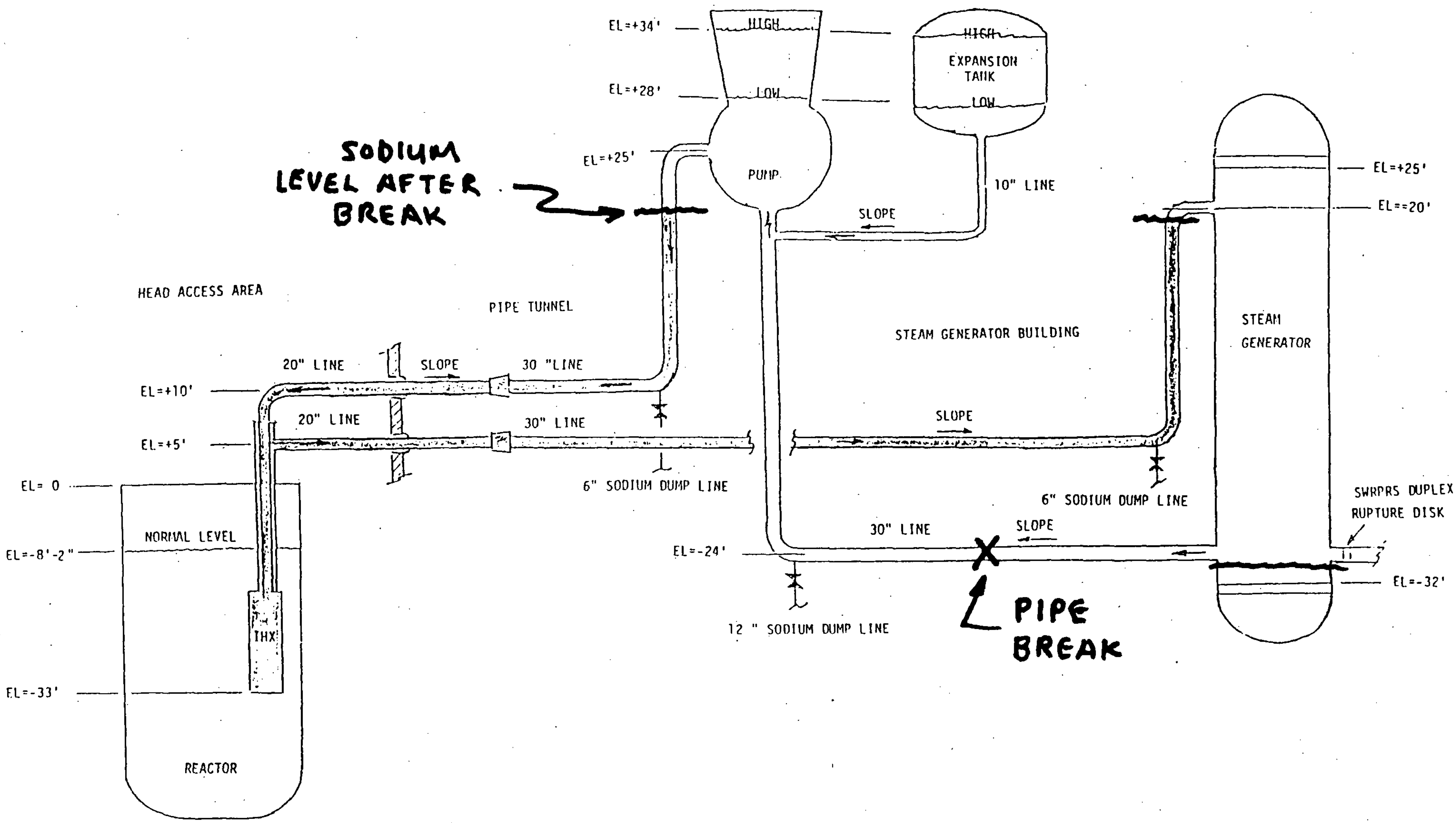
A major pipe break in the 30 inch diameter steam generator sodium inlet line would result in a sodium spill of ~22,000 gallons. The pipe break location and sodium levels in the system after drainage are shown on the IHTS Hydraulic Profile drawing in Figure E.8-20. The break occurs at elevation +5 feet and drains the IHTS pump, expansion tank and upper portion of the steam generator as well as the IHX inlet/outlet lines. The sodium-air interface in the IHX inlet/outlet lines is in the HAA directly above the IHX nozzles after the break. About 2000 gal. of sodium remain in each of the IHX units and the vertical 20-inch diameter inlet and outlet piping. This is considered to be the most severe condition because of the proximity of the sodium-air interface to the IHX. With the reactor shut-down and loss of the IHTS, reactor decay heat removal will be through the RVACS. The reactor system will heatup to 1200°F, and the intermediate sodium expansion will push about 200 gallons of sodium out of each IHX riser into the horizontal IHTS lines and out the pipe break. During the cooldown of the reactor to 400°F, the intermediate sodium in the IHXs contracts about 100 cu. ft. and the sodium-air interface drops about ten feet below the reactor deck to a level one foot above the IHX upper plenum.

d) Corrective Action

After drainage of the IHTS, following a pipe break, air will enter the open pipe at the break and continue to react with the residual sodium. If access is available to the break site, the break could be covered with a gas tight barrier to prevent further reaction; however, it can not be assumed that the steam generator building will be accessible shortly after a major sodium fire. Oxygen in the air entering the pipes after sodium drainage will react until the air is exhausted of oxygen and only an inert blanket of nitrogen is left at the sodium interface. In a confined space, such as that in the piping, the sodium-oxygen reaction rate will be very slow (less than two pounds per hour per square foot) and should essentially cease when the oxygen content in the piping drops and the sodium freezes.

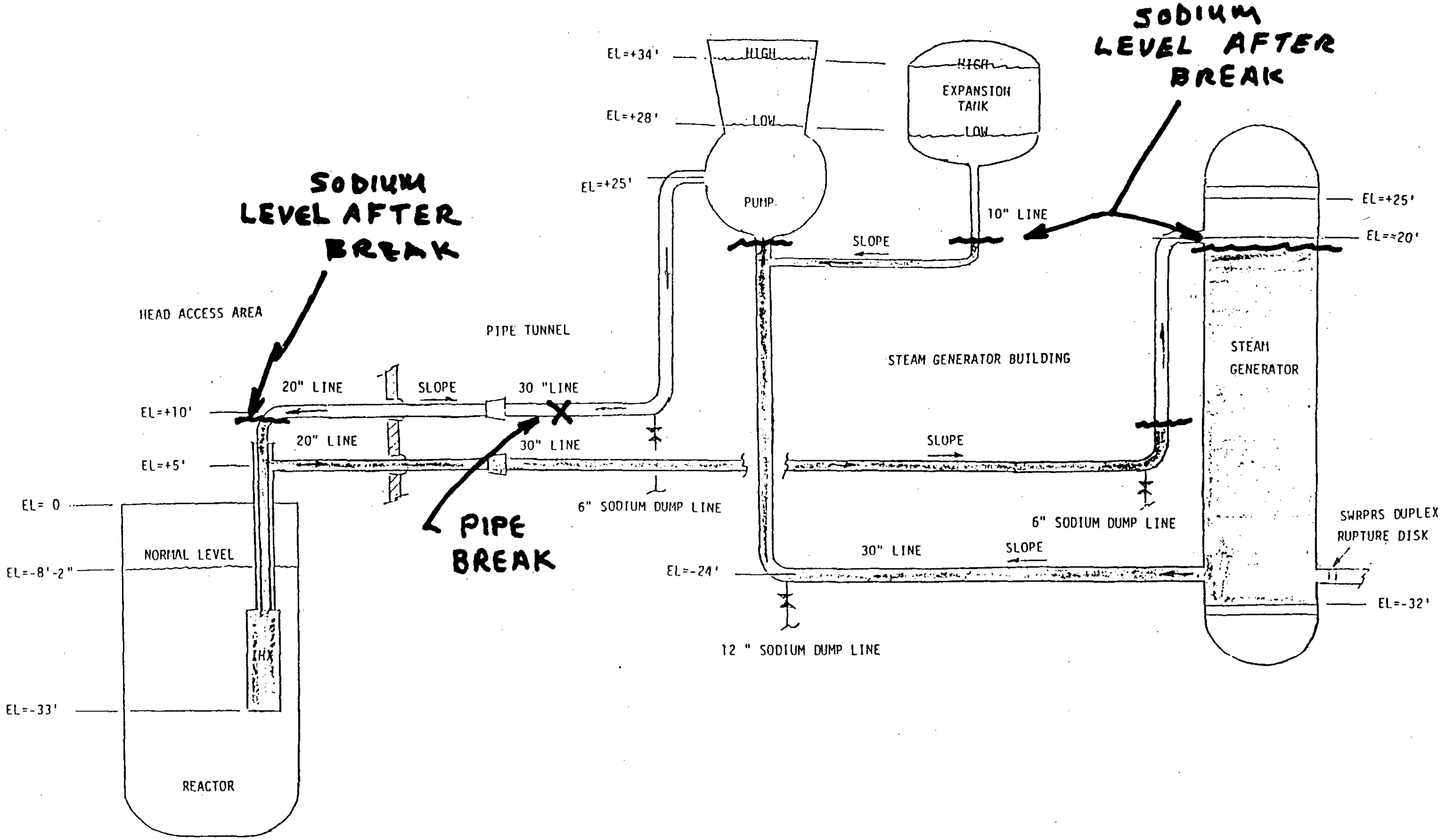
The maximum temperature at the reaction zone with an adequate air supply is estimated to be 1100 - 1300°F.

Since the IHTS piping within the HAA has a guard pipe to protect against breaks and the HAA is isolated from the steam generator building, it is assumed that the pipe break occurs in the steam generator building (or pipe tunnel) and that the HAA is accessible at all times. Corrective action would consist of removing the thermal insulation and guard piping from a 15 foot length of each pipe near the HAA penetration (away from the IHXs) and allowing the pipes to cool. A sodium filled line would cool down from the maximum RVACS temperature of 1200°F to near ambient temperature and form a sodium freeze seal in less than 24 hours by natural convective heat transfer to the air in the HAA. If a line did not contain sodium, the cool down time would be considerable shorted. External cooling could also be used to reduce the cooldown time. The four 20 inch IHX inlet/outlet lines would then be mechanically cut and inerted. A short section (2-3 feet) of each line would be removed and the ends capped with welded fittings to provide a positive air barrier.



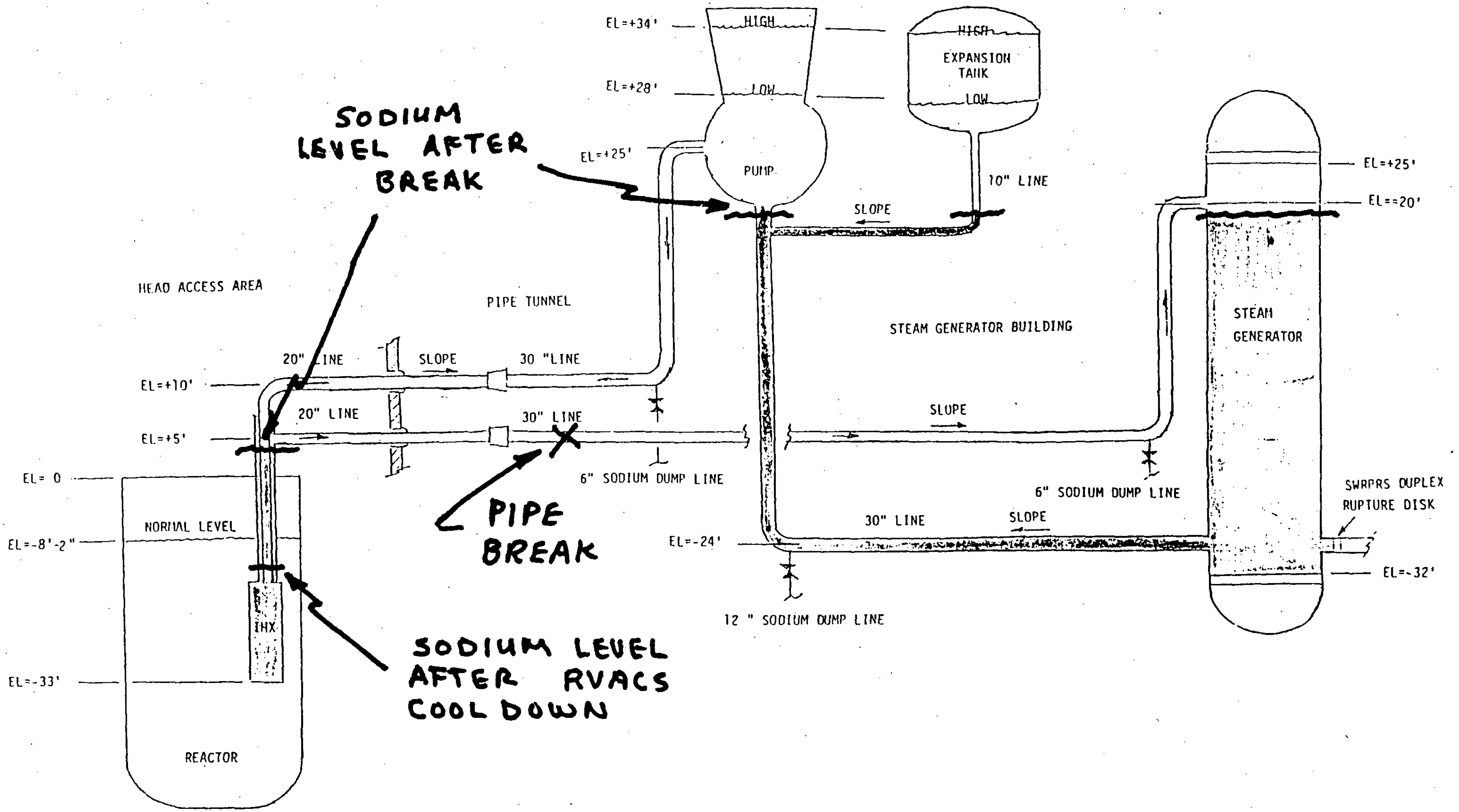
IHTS HYDRAULIC PROFILE

Figure E.8-18



IHTS HYDRAULIC PROFILE

Figure E.8-19



IHTS HYDRAULIC PROFILE

Figure &E.8-20

Event 6b Reactor Vessel (RV) Leak (Critical Leak)

Sodium leaking from the reactor will be contained by the containment vessel. The containment vessel is sized to maintain a 5 inch (nominal) annulus around the reactor vessel which minimizes the sodium level drop during a leak. At the final sodium leak level (after maximum volume of sodium leakage), the normal reactor flow paths and in-vessel spent fuel cooling are maintained; that is, the sodium is still above the IHX inlets and above the stored spent fuel.

Sodium leak detection is provided by four sodium level detectors located in the reactor vessel and by three sodium aerosol detectors and three sodium liquid detectors located inside the containment vessel. The sodium level detectors are sensitive to within two inches. A level drop of two to five inches will be indicated to the operator via alarms and result in a normal reactor shutdown. A level drop of six inches or more will cause a reactor scram since the level detectors are part of the reactor protection system. Very small leaks (pin hole size) will generate aerosols and be detected by the aerosol detectors in the containment vessel. Sodium accumulating at the bottom of the containment vessel will be detected by contact type detectors. Indication from either of these will cause the reactor to be shut down.

Performance of the primary and intermediate heat transport systems are unaffected by a leak in the reactor vessel. Since the IHX inlets are covered by sodium at the final leak level, the reactor primary pumps and IHXs remain functional for transporting the core heat to the IHTS and on to the turbine generator. Reactor vessel auxiliary cooling system (RVACS) performance is improved by reactor vessel leakage since replacing the argon with sodium between the two vessels improves heat conduction from the reactor vessel to the containment vessel. In the event the IHTS is not available, the RVACS will remove the core decay heat. With RVACS only cooling but without a reactor vessel leak, the reactor sodium temperature will peak at 1108°F after 26 hours. The same event with reactor vessel leakage results in a peak reactor sodium temperature of 971°F or 137°F lower than occurs without reactor vessel leakage.

The size and location of the leak in the reactor vessel has no adverse impact on the performance of the heat transport system and the decay heat removal systems. A relatively small leak will require many hours or days to reach the final leak level. This will give the operator ample time to shut down the reactor following normal procedures. A large leak will cause the reactor sodium level to decrease more rapidly and cause the reactor to be shut down by fast runback means or by scramming the control rods. For all range of leakages the final leak level in the reactor remains above the IHX inlets.

D. Radiological Dose Assessment

Of the proposed Bounding Deterministic Events analyzed in the preceding section, only Events 1b and 3 lead to fuel failures. Thyroid and whole body doses at the site boundary were estimated for Bounding Events 1b and 3. Two exposure times were analyzed: 36 hours and 30 days. The dose estimates include contributions from external radiation by the passing cloud and internal radiation caused by the inhalation of radionuclides. The inhalation dose was based on the total internal dose accumulated over 50 years after the accident. Conservative dispersion factors, breathing rates, and dose conversion factors were used to estimate the dose. The calculated exposure was based on the following assumptions.

1) Radioactivity Inventory: End-of-equilibrium cycle (EOEC) radioactivity inventory of the PRISM 1986 metal core was used for the dose assessment. The inventory was estimated using the ORIGIN computer code.

   A 20-month irradiation cycle at 85% equivalent availability was assumed (91.3 shutdown days followed by 517.4 equivalent full power days). The EOEC core has the following irradiation history.

   a) Driver fuel (42 assemblies):
      14 assemblies irradiated for 3 cycles,
      14 assemblies irradiated for 2 cycles, and
      14 assemblies irradiated for 1 cycle.

   b) Internal blanket (25 assemblies):
      12 assemblies irradiated for 2 cycles, and
      13 assemblies irradiated for 1 cycle.

   c) Outer blanket (36 assemblies):
      12 assemblies irradiated for 5 cycles,
      12 assemblies irradiated for 4 cycles, and
      12 assemblies irradiated for 3 cycles.

2) Source Term Basis: Table E.8-2 presents the key source term parameters used in the dose assessment of Bounding Events 1b and 3. The table shows the fraction of clad damaged, fraction of fuel molten, time and duration of release of radioisotopes from the fuel assemblies to the primary sodium coolant, and the containment leak rates. The values of these parameters shown in Table E.8-2 have been derived from the event analyses discussed in Section C. Table E.8-2 also indicates that no credit was taken in the dose assessment for the holdup or attenuation in the head access area or release paths.

3) Fraction of Radioisotopes Released from Fuel to Primary Na: Table E.8-3 presents the fraction of radioisotopes released from the damaged fuel elements to the primary Na for Bounding Events 1b and 3. As seen in the table, 100% of the noble gases in the damaged fuel elements was assumed to be released. For other radionuclides, the WASH-1400 (Reference 1) release fractions were used for solid fuel (90% of the damaged fuel elements in Event 1b) and molten fuel (10% of the damaged fuel in Event 1b and 100% of the damaged fuel in Event 3).

4) Fraction of Radioisotopes Released from Primary Na to Cover Gas Region
Table E.8-4 presents the fraction of radioisotopes vaporized from primary Na to the cover gas region. The fractions were estimated using the Rayleigh equation for equilibrium vaporization (Reference 2):

$$F_r = 1 - (1-F_{Na})^{A_r}$$

where

F_r = Fraction vaporized of radioisotope r
F_{Na} = Fraction of Na vaporized
A_r = Temperature-dependent characteristic constant for the vaporization of radioisotope r from Na

The values of A_r were estimated from Reference 2 for the Na temperatures of Bounding Events 1a and 3 shown in Table E.8-4. The fraction of Na vaporized (F_{Na}) was estimated from the Na vapor pressure at these temperatures and the volume of the cover gas region. As seen in Table E.8-4, the primary sodium has a significant capability to retain non-gaseous radioisotopes as long as the Na vapor is maintained at the indicated small fraction. The table also shows that next to noble gases, the alkali metals (e.g., Cs) are least retained by sodium.

5) Atmospheric Dispersion: Atmospheric dispersion (X/Q) factors at the site boundary (0.5 mile) were obtained from Reg. Guide 1.4.

6) Dose Conversion Factor: Dose conversion factors from Reg. Guide 1.109 and NUREG/CR0150 were used in this assessment. For the inhalation dose, 50-year dose commitment conversion factors were used.

The estimated 36-hour and 30-day doses for Bounding Events 1b and 3 are shown in Table E.8-5. The table presents the lower PAG limits for thyroid and whole body doses. The results in the table lead to the following conclusions.

1) The doses for Events 1b and 3 are well within the lower PAG dose limits.

2) The whole body dose is more limiting than the thyroid dose. In all cases, the thyroid dose is <1% of the lower PAG limit. On the other hand, the whole body dose reaches 8% of the PAG lower limit for the 30-day dose of Event 3. The low level of the thyroid dose is attributed to the strong capability of the primary sodium to retain the iodine isotopes (which present maximum hazard to the thyroid per Curie inhaled) and the Cs isotopes (which are next to iodine in their hazard to the thyroid). The whole body dose, on the other hand, is dominated by the external radiation from the noble gases which are not attenuated by the primary sodium.

3) For Bounding Event 1b, the whole body dose is not significantly sensitive to the exposure time after the first 36 hours (the 30-day whole body dose is only 20% larger than the 36-hour dose). This is due to the decay of the noble gases which constitute most of this dose before the release begins.

4) The 36-hour whole body dose of Event 3 is less than the corresponding dose of Event 1b, despite the larger leak rate and fuel damage of Event 3. This is attributed to the fact that fuel failure in Event 3 is delayed to 22 hours while the fuel-failure delay for Event 1b is only two hours after the accident initiation. This difference in release delay allows for the decay of the short-lived fission gas. As the exposure increases to 30 days, however, the long-lived Cs isotopes become dominant contributors to the whole body dose of Event 3. This leads to an order of magnitude increase in the dose from its 36-hour value of 7.5×10^{-3} Rem to the 30-day value of 8×10^{-2} Rem. Despite this significant increase, the 30-day dose for Event 3 is only 8% of the PAG lower limit for whole body dose.

REFERENCES:

1. Reactor Safety study, WASH-1400 (NUREG-75/014), Appendix VII, "Release of Radioactivity in Reactor Accidents," USNRC, October 1975.

2. A. W. Castleman, Jr., "LMFBR Safety, I. Fission-Product Behavior in Sodium," Nucl. Safety, Vol. II, No. 5, September-October 1970.

TABLE E.8-2 - SOURCE TERM BASIS

Bounding Event Number	1b	3
Core Fraction Involved		
Driver Fuel	0.286	1.0
Inner Blanket	0.0	1.0
Radial Blanket	0.0	1.0
Molten Fuel Fraction	0.1	1.0
Time of Release From Fuel (Hours After Accident Initiation)	2.0	22.0
Duration of Release From Fuel (Hours)	14.0	1.0
Containment Leak Rate %/Day		
0-36 Hours	0.1	0.3
36 Hours - 30 Days	0.1	0.1
Head Access Area and Release Path Holdup or Attenuation	None	None

TABLE E.8-3

FRACTION OF RADIOISOTOPES RELEASED FROM FUEL TO PRIMARY Na

Bounding Event	1b	3
Noble Gases	1.0	1.0
Halogens	0.12	0.9
Alkali Metals	0.15	0.8
Sr, Ba	0.01	0.1
Others	0.0003	0.003

TABLE E.8-4

FRACTION OF RADIOISOTOPES RELEASED FROM PRIMARY Na TO COVER GAS REGION

Bounding Event	1b	3
Primary Na Temperature °K	980 (1305°F)	1150 (1610°F)
Fraction Vaporized From Primary Na to Cover Gas		
Na	$<10^{-5}$	$5x10^{-5}$
Noble Gases	1.0	1.0
Halogens	$3x10^{-6}$	$2x10^{-5}$
Alkali Metals	$2x10^{-4}$	$5x10^{-4}$
Sr	$3x10^{-7}$	$3x10^{-6}$
Ba	$2x10^{-8}$	$2x10^{-7}$
Te	$<10^{-8}$	$2x10^{-8}$
Others	$<10^{-8}$	$<2x10^{-8}$

TABLE E.8-5

SITE BOUNDARY DOSE ESTIMATES

		Bounding Event							
		1b				3			
		36 Hr		30 Day		36 Hr		30 Day	
Organ	PAG Limits (Rem)	Dose (Rem)	% PAG	Dose (Rem)	% PAG	Dose (Rem)	% PAG	Dose (Rem)	%PAG
Thyroid	5.0	1.1 E-4	0.002	2.9 E-4	0.006	1.4 E-2	0.28	4.3 E-2	0.86
Whole Body	1.0	5.1 E-2	5.1	6.1 E-2	6.1	7.5 E-3	0.75	8.0 E-2	8

ADDENDUM A. TOLERANCE OF RVACS TO BLOCKAGES

A number of beyond design basis cases have been analyzed to demonstrate the performance of the RVACS under faulted conditions and the system's exceptional tolerance to flow blockages. These results are summarized in Table E.8-A1.

TABLE E.8-A1. SUMMARY RESULTS FOR RVACS BLOCKAGE CASES
(Reactor Scrammed, Decay Heat Removal by RVACS Only)

	Max Avg Core Sodium Outlet Temperature (F)
Nominal Operation	1108
Three (of four) air inlets blocked	1113 (+5)
Three (of four) air outlets blocked	1116 (+8)
Three air inlets and three air outlets blocked	1120 (+12)
All air inlets blocked or total flow blockage at bottom of RVACS	1155 (+47)
10x nominal air flow path resistance	1180 (+72)
100x nominal air flow path resistance	1410 (+302)

Before these results are discussed in more detail, a short summary of the calculational method will be presented.

A.1 Analysis Model

The same basic one-dimensional calculational model was used for the beyond design basis analyses as for the nominal analysis. The computer code used is a condensed version of CINDA* which runs on a VAX-750 computer. It utilizes a lumped-parameter thermal network representation of the physical systems and solves the resulting mathematical description using implicit finite difference numerical techniques.

The input parameter assumptions used are summarized in Table E.8-A2. The core decay heat curve is calculated for end of life equilibrium cycle conditions using the computer code ORIGEN2. Heat generation from the stored fuel was also included by conservatively assigning a constant heat generation value obtained from the decay heat curve at 23 days (time required for refueling following reactor shutdown). Air-side heat transfer coefficients for the RVACS hot air riser are based on recent data from the ANL air-side RVACS

* Lewis, D.R., et al., Chrysler Improved Numerical Differencing Analyzer for 3rd Generation Computers," Chrysler Space Division, TN-SP-67-287, October 1987.

tests; because of a strong entrance effect, the data are correlated as a local Nusselt number vs local bulk Graetz number:

$$Nu,f = 6.036\ Gz,b^{0.314} \qquad (A-1)$$

The thermal emissivities of the oxided 316SS and 2-1/4Cr1Mo vessels have been experimentally determined and fitted by the following correlation:

$$E = 0.690 + 8.0 \times 10^{-5} T(°F) \qquad (A-2)$$

TABLE E.8-A2 INPUT PARAMETER ASSUMPTIONS

Decay heat curve	Calculated for 1987 metal core at EOEC
Heat transfer coefficient	Nominal ANL data, eq. A-1
Thermal emissivity	Eq. A-2, 0.77 at 1000°F
Bottom head heat loss	Projected surface area effective
IHTS heat loss	0
Reactor head heat loss	0

The effect of heat losses from the bottom of the reactor assembly was calculated by using the projected area of the bottom head as effective for heat transfer. It was also assumed that the heat losses through the IHTS and the reactor head were zero at all times. The daily average RVACS air inlet temperature was assumed to be 100°F. The flow resistances of the primary sodium and air flow paths were calculated for the various cases using approaches which are considered conservative.

A.2 Analysis Results

As shown in Table E.8-A1, the RVACS is extremely tolerant of various types of flow blockages.

This tolerance to major blockages is the result of two major factors. First, the majority of the thermal resistance in the system is in the argon-filled gap region where heat transfer is dominated by thermal radiation and is not affected by the air flow rate. Second, a reduction in the air flow rate, e.g., by blockages, will be partly offset by an increased air temperature rise. For example, for the case with three inlets and outlets blocked (75% total blockage), the air flow rate decreased from 49.5 lb/sec to 39.1 lb/sec and the air temperature rise increased from 211°F to 262°F. The resulting thermal performance was only reduced slightly primarily due to a lower convective heat transfer coefficient.

In order to evaluate the case where all air inlets are completely blocked, several assumptions subject to experimental verification were made. One of these is that air trapped in the hot air riser will not remain there when heated sufficiently and will rise to the outlet plenum. In so doing, colder air from the outlet plenum will have to replace the hotter air escaping since a partial vacuum condition can not exist in the hot air riser. Thus, it is postulated that preferential downflow and upflow zones are created in which about one-half of the hot air riser cross-sectional area has downflow while the other half has upflow as indicated schematically in Figure E.8-A1. The downflow zone is visualized to be located in the two colder quadrants where the electromagnetic pumps are positioned while the upflow zones are visualized to be in the hotter IHX quadrants. The flow makes a U-turn in the gap region near the bottom of the vessel as indicated in Figure E.8-A1. It is further assumed that cold air downflow is established in two of the four air outlet stacks while the hot air upflow is in the remaining two air outlet stacks. Establishment of flows in the outlet stack and the flow pattern will depend on what thermal unbalances and external (wind) conditions exist initially.

Results of the analysis for this postulated case using the U-airflow model described above show that the maximum core sodium outlet temperature increases to 1155°F which is well below the design basis temperature limits of 1200°F (Level C) and 1300°F (Level D). The analysis shows that the performance is not sensitive to mixing of hot and cold air that undoubtedly occurs at the interfaces of the postulated down and upflow zones. The air flow rate at peak RVACS performance was 231.4 lb/sec and the air temperature rise was 459°F. About one-half of the air temperature rise occurred in the downflow air stream. Further detailed analytic and experimental work is required to verify the U-airflow model assumptions.

It is important to note that the case of all air inlets blocked is equivalent to a total blockage at the bottom of the RVACS in the reactor silo. That is, if sufficient sand or mud were to enter the RVACS to fill the silo up past the bottom of the collector cylinder, the effect would be identical to that of all inlets blocked with a U-flow natural air circulation being initiated in the hot air riser annulus.

An alternative way to express the RVACS tolerance to blockages is in terms of percent increase in the air flow path resistance. This can be considered to represent an extended region of blockage, such as rubble or loose debris filling part of the vertical flow passages. As shown in Figure E.8-A2, up to about 60 times the nominal air flow resistance can be tolerated without exceeding the Level D limit within 36 hours.

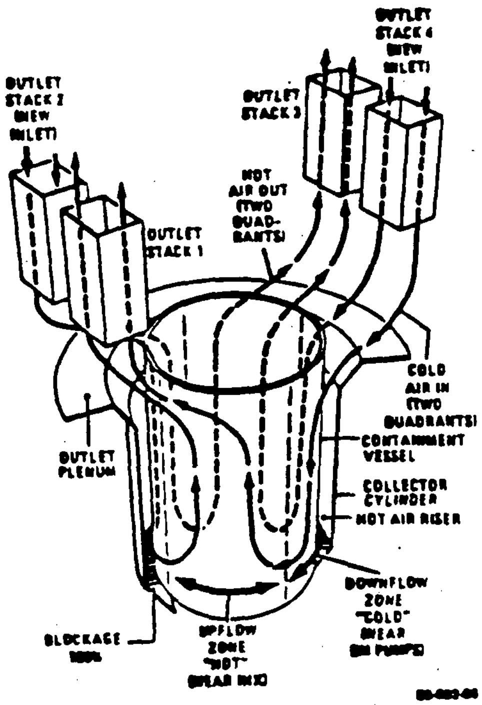


Figure E.8-A-1 U-Air Flow Model for Natural Convection Flow Pattern in RVACS Hot Air Riser with 100% Blocked Air Inlet

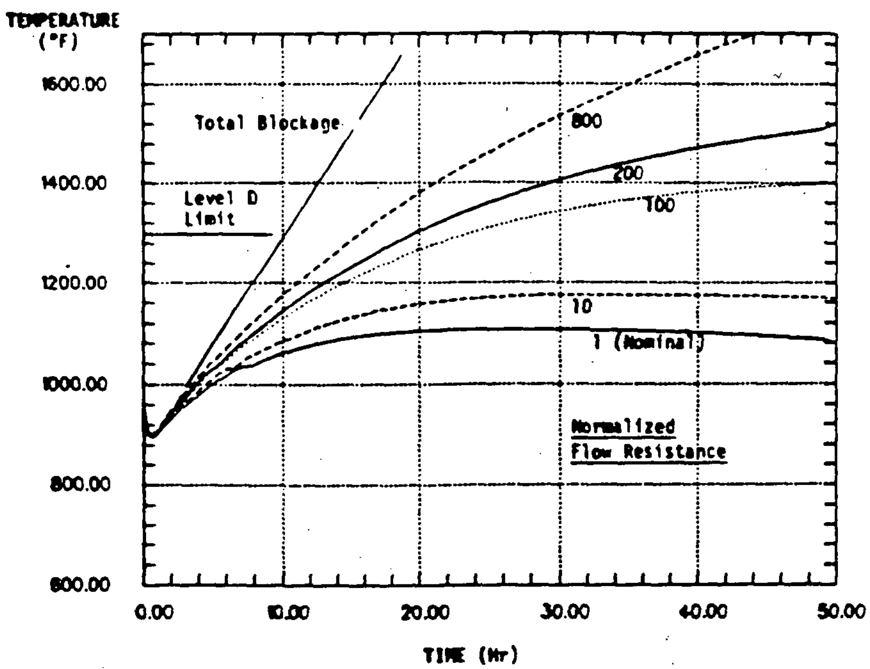


Figure E.8-A-2 Effect of RVACS Air Flow Path Resistance on Average Core Sodium Outlet Temperature

E.9 Comment

Provide an assessment and documentation of enhanced safety characteristics and margins of the PRISM conceptual design regarding:

- o Long response time
- o Reduced potential for operator error
- o Capability to retain fission products
- o Highly reliable safety systems (passive/inherent characteristics)
- o Simplification (systems/analysis)

Response

A major goal in the PRISM design effort has been to incorporate the safety attributes listed in the NRC's Regulation of Advanced Nuclear Power Plants, Statement of Policy, 10 CFR Part 50 as announced in the Federal Register, Vol. 51, No. 130 on July 8, 1986. The nine attributes listed in the Statement of Policy are summarized below:

1. Highly reliable and less complex shutdown and decay heat removal systems using inherent or passive means

2. Longer time constants

3. Simplified safety systems, reduced operator actions

4. Minimization of potential for severe accidents by providing sufficient inherent safety, reliability, redundancy, diversity and independence in safety systems

5. Reliable BOP equipment, safety system independence from BOP

6. Easily maintainable equipment and components

7. Reduced personnel exposure

8. Defense-in-depth against fission product release

9. Design features that are, or can be, proven

This response is organized to address the nine attributes listed above, but in the order listed in Comment E.9.

Long Response Time - (Attribute 2) - In evaluating this question, the time to reach each of the following five limits was evaluated:

Fuel failure
Sodium boiling
Safety structural failure
Protective Action Guidelines (PAG) limits
10CFR100 limits

These evaluations show that the PRISM reactor is designed to be able to withstand indefinitely all Design Basis Events (DBE) and Beyond Design Basis Events (BDBE) discussed in Chapter 15 and Appendix E of the PSID. Therefore, for these events, ample time is available for operator action to alleviate the causes and consequences of these events. Table E.9-1 summarizes these evaluations.

In addition to the DBE and BDBE discussed in the PSID, the NRC has requested evaluation of seven additional Bounding Deterministic Events intended to bound the DBE and BDBE spectrum. (See Comment E.8). For some of these events, a limit is reached. Table E.9-1 presents the results for all but the external events which have not yet been defined in sufficient detail by the NRC to perform an evaluation.

Only two events exceed the limits. Unprotected withdrawal of all control rods, assuming reactor vessel auxiliary cooling system (RVACS) only with coincident, instantaneous loss of all IHTS cooling, leads to initiation of fuel failure after two hours in the 12 inner fuel assemblies. The probability of this event is so low as to put it in the probabilistic risk assessment (PRA) category. Nevertheless, there is no sodium boiling, no structural failures, and no exceeding PAG or 10CFR100 levels.

The other event which leads to exceeding a limit is the assumed loss of all decay heat removal capability by the steam system, by the auxiliary cooling system (ACS), and by RVACS for the first 36 hours at which time 25% unblockage is assumed, but assuming the system is scrammed. This event leads to fuel failure in 17 hours. The probability of this event is so low as to put it in the PRA category. Nevertheless, there is no sodium boiling, no structural failures, and no exceeding PAG or 10CFR100 levels. All failures in this event can be prevented if only 10% of the RVACS cooling capability can be restored during the first ten hours into the transient. A more detailed discussion of this event, and the improbability of complete loss of RVACS, are presented in Comment E.8.

In addition to the DBE, BDBE, and Bounding Deterministic Events discussed in Table E.9-1, additional improbable events have been evaluated which fall into the Probabilistic Risk Assessment (PRA) category. These events are discussed in Appendix A in the PSID. They assume energetic core disruptions which lead to core melt, vessel rupture, and reactivity release. A review of these events shows the following time ranges to sodium boiling, core uncovering, and sodium depletion:

Sodium boiling	-	15-25 hours
Core uncovering	-	26 - 99 hours
Sodium depletion	-	87 - 124 hours

(The data in Appendix A currently show sodium boiling occurring after five hours for several events. However, these early studies were extremely conservative, and at least for sodium boiling greatly underestimate the time available. Based on more recent analyses, at least 15 hours are available in the worst case before sodium boiling initiates). Thus, even for PRA events, there is significant time for actions to alleviate the consequences of these improbable events.

Table E.9-1

RESPONSE TIMES FOR VARIOUS EVENTS

	TIME TO LIMITS				
DBE	FUEL FAILURE (hours)	SODIUM BOIL (hours)	SAFETY STRUCTURAL FAILURE (hours)	PAG LIMITS (hours)	10CFR100 LIMITS (hours)
FAST RUNBACK	*	*	*	*	*
SCRAM	*	*	*	*	*
LOSS OF NORMAL SHUTDOWN COOLING	*	*	*	*	*
LOCAL FAULTS	*	*	*	*	*
SODIUM SPILLS	*	*	*	*	*
FUEL HANDLING & STORAGE	*	*	*	*	*
COVER GAS RELEASE	*	*	*	*	*
BDBE					
UNPROTECTED LOSS OF FLOW	*	*	*	*	*
UNPROTECTED TRANSIENT OVERPOWER	*	*	*	*	*
UNPROTECTED LOSS OF HEAT SINK	*	*	*	*	*
UNPROTECTED 6-ROD TRANSIENT OVERPOWER	*	*	*	*	*
NRC DETERMINISTIC EVENTS					
UNPROTECTED WITHDRAWAL OF ALL CONTROL RODS FOR 36 HOURS (with forced cooling)	*	*	*	*	*
UNPROTECTED WITHDRAWAL OF ALL CONTROL RODS FOR 36 HOURS (RVACS only)	>2	*	*	*	*
STATION BLACKOUT FOR 16 HOURS WITH LOSS OF AC POWER	*	*	*	*	*
LOSS OF FORCED COOLING, LOSS OF ACS, LOSS OF RVACS, WITH SCRAM AND 25% RVACS UNBLOCKAGE AFTER 36 HOURS	>17	*	*	*	*
INSTANTANEOUS LOSS OF FLOW FROM ONE PRIMARY PUMP	*	*	*	*	*
STREAM GENERATOR TUBE RUPTURE WITHOUT ISOLATION OR WATER DUMP	*	*	*	*	*
LARGE SODIUM LEAK	*	*	*	*	*
EXTERNAL EVENTS	N/A	N/A	N/A	N/A	N/A

NOTE: * means reactor conditions stabilize, and limits are never reached.

<u>Reduced Potential for Operator Error - (Attributes 3 and 6)</u> - Attribute No. 3 pertains to simplified safety systems. PRISM safety systems are simplified, compared to traditional nuclear power plant safety systems, in three major ways. First, there are fewer safety systems. Second, the few safety systems that are included are, with one exception, inherent shutdown mechanisms or passive heat removal systems, requiring no operator action and providing immunity to operator error. Third, the one active safety system, the Reactor Protection System (RPS), is completely automatic, requires no operator action, and is designed to provide immunity to operator error. Thus, minimal safety action demands are placed on the operators, and built-in design features prevent operator error from impeding the performance of the safety systems.

In addition to the above simplified safety features, the plant information system is designed to present data to the operators in a well processed, simple format, for easy understanding of plant status. Any plant disturbance is evaluated using real-time and process models, and decision aids and prompts are provided to the operator in real time. Trends are displayed along with the current value for easy understanding of the plant status. Capability of analyzing historical (archived) and sequence of events data is provided for evaluation of plant transients. An integrated alarm system is provided which analyzes alarms, and presents them in order of importance so that in the event of an accident the operator is not flooded with less important alarms.

Attribute No. 6 pertains to easily maintainable equipment and components. Such equipment and components reduce the potential for operator error. The major contributor to this goal is the reduced number of systems in PRISM, and hence the reduced number of equipment and component items requiring maintenance. For example, the primary control and safety systems on PRISM are the plant control system (PCS), the reactor protection system (RPS), the reactor vessel auxiliary cooling system (RVACS), and seismic isolation, a total of four. The comparable systems on a light water reactor are the PCS, RPS, two high pressure coolant injection systems, two low pressure coolant injection systems, a reactor pressure vessel pressure relief system, a containment isolation system, and a containment pressure suppression system, a total of nine. Thus, there are fewer systems, equipment items, and components for the operators to be cognizant of and to maintain. In addition, since there are nine reactors per power plant with rotating outages, the plant can justify a permanent well trained maintenance staff, performing frequent and repetitive maintenance to become experts in the plant equipment and components. Also, since there are nine reactors per power plant, if one reactor goes down, the plant can continue to produce 8/9 ths of its rated power. Thus, there is less pressure to rush maintenance activities, permitting a more orderly maintenance schedule less prone to errors and oversights.

<u>Capability to Retain Fission Products - (Attributes No. 8 and 7)</u> - Attribute No. 8 addresses the defense-in-depth concept to ensure safety through prevention, protection and mitigation. A sub-set of this concept addresses the capability to retain fission products. This capability can be evaluated in three ways. The first is the succession of physical barriers between the fission products in the fuel and the external environment. The second is the large inherent safety margins in PRISM which

minimize challenges to the physical barriers. The third is the safety systems which prevent or mitigate the severity of any accident.
The succession of physical barriers includes the cladding surrounding each fuel pin (with a fission gas plenum to alleviate pressure buildup), the reactor vessel surrounding the core, and the containment vessel surrounding the reactor vessel. The reliability of these barriers is enhanced by having no penetrations in the reactor vessel or containment vessel, by having hermetically sealed joints and penetrations in the vessel closure head, by operating at near atmospheric pressure, by locating the reactor module below ground, by learning curve improvements due to the many units required, by enhanced quality assurance due to factory module fabrication and control, by large margins to failure, and by continuous monitoring for fuel failures and vessel leaks. Adding to the public protection of these physical barriers are the exclusion zone around the plant, and the long time available (over 36 hours) to evacuate the public if required.

The large inherent margins designed into PRISM minimize the number of challenges to the system and to the physical defense-in-depth barriers. Among the most important margins are the margins to fuel failure, seismic margins, and margins to high thermal stresses. Large margins to fuel failure are maintained in order to provide time for the plant control system runback capability, and the reactor protection system, to operate to protect the core without fuel life degradation, and for inherent core responses to terminate unprotected transients without fuel failures. The minimum margin to centerline fuel melting occurs at the time of minimum fuel conductivity, approximately 6 months into the first operating cycle. At this time, the nominal margin to centerline fuel melting is 245°F. Including the effects of hot channel factors and the uncertainty in fuel solidus measurements, the 2 sigma margin to centerline fuel melting is 95°F. The corresponding minimum temperature margins for the fuel/clad eutectic cladding attack are 196°F and 89°F. With these margins, the number of fuel pin failures is predicted to not exceed 0.11 per cycle with 95% confidence. In addition, as shown in Table E.9-1, fuel failures are predicted for none of the design basis or beyond design basis events, and for only two of the deterministic spectrum bounding events.

The seismic and thermal margin issues are related but in inherent conflict for conventional design, since thick components are desired for high seismic margins, and thin components are desired for high thermal margins. These two conflicting demands are decoupled in the PRISM design by use of seismic isolation and a vertically stiff reactor vessel. Horizontal seismic isolation bearings support the nuclear safety related equipment, and prevent horizontal ground motion from significantly influencing the design, while the small diameter, vertically stiff reactor vessel limits deflections and stresses due to vertical ground motion.

Thermal stresses in the reactor vessel are low due to fewer, less severe thermal transients, and to the ability to use thinner components since the vessel operating pressure is essentially atmospheric and the seismic forces are low. The features which minimize thermal transient effects are:

- o Reactor vessel size, shape, and arrangement of internals
- o One steam generator for one reactor vessel
- o Elevated steam generator with steam drum and recirculation piping

The small diameter reactor vessel confines the coolant flow path, and allows accurate prediction of flow paths and thermal gradients. As mentioned above, the low operating pressure of PRISM allows thin wall construction, reducing thermal stresses. Vertical stacking of the internal heat exchangers (IHX) above the core increases thermal center differences, maximizing natural circulation of the primary coolant, and ensuring flow over a wide range of normal and upset conditions. The elevated steam generator and steam drum provide for natural circulation in the intermediate heat transfer system, further assuring natural circulation in the reactor vessel. Symmetrical location of the IHXs, and the use of four electromagnetic pumps, minimize thermal transient effects due to loss of one pump. The use of one steam generator per reactor minimizes thermal transients when the steam generator is lost since asymmetric flow paths to a second steam generator are not possible.

The capability to retain fission products, and maintain the margins discussed above, are further enhanced by a number of safety features and systems, both passive and active. The primary passive features and systems are a strong negative reactivity feedback coefficient which accommodates all unprotected thermal overpower, unprotected loss of flow, and unprotected loss of heat sink transients; the auxiliary cooling system (ACS) which removes decay heat through the intermediate heat transfer system; and the passive safety grade reactor vessel auxiliary cooling system (RVACS) which removes decay heat by natural air circulation around the containment vessel. The primary active safety system is the reactor protection system (RPS). The RPS needs to monitor only five primary reactor operating parameters, due to the inherent simplicity of the PRISM design. These five are neutron flux, pressure (flow), core inlet temperature, core outlet temperature, and coolant level in the reactor vessel. The RPS is a Class 1E, four division system that is fully automatic in the operation mode, fault-tolerant, and automatic in self test and calibration modes. Its signals command reactor scram, both by gravity drop and motor drive-in of the control rods; primary system flow coastdown; and safe shutdown monitoring. Supporting the RPS is the control rod runback feature of plant control system (PCS), which serves to limit transients before they reach levels that would activate the RPS.

The capability to retain fission products under both normal and transient conditions contributes to the benefit of low personnel exposure, Attribute No. 7 in the Statement of Policy. In addition, the pool design of PRISM keeps the activated primary sodium inside the reactor vessel. This feature, coupled with the below ground location of the reactor module, minimizes personnel exposure in the intermediate heat transfer system and balance of plant (BOP) areas. The lack of penetrations in the reactor vessel, the atmospheric operating pressure, and the hermetically sealed vessel closure head, virtually eliminate any release of radioactive gas into the environment under operating conditions. During shutdown, when the hermetic seals are broken, the sealing function is performed by the refueling cask. The result of all these features is a predicted maximum whole body exposure of less than 20 man-Rem/year/reactor module.

Highly Reliable Safety Systems (Passive, Inherent Characteristics) - (Attributes Nos. 1, 4 and 5) - Attribute No. 1 pertains to highly reliable, simple, inherent, and passive shutdown and decay heat removal systems. The primary inherent shutdown mechanism for PRISM is the strong negative reactivity feedback coefficient which accommodates all unprotected thermal overpower, unprotected loss of flow, and unprotected loss of heat sink transients. This feature is augmented by a low excess reactivity content in the core due to its near zero burnup reactivity change over its operating cycle, which translates into reduced control rod worth requirements and low overpower potential in rod fault events; low power density permitting a large power surge before cladding damage occurs; and low operating temperatures permitting a large power surge before sodium boiling occurs.

The primary passive decay heat removal system is the reactor vessel auxiliary cooling system (RVACS) which is always in operation, removing minimal heat under normal conditions, but removing all decay heat when normal heat rejection systems are lost. It has no moving parts, and requires no automatic or operator actions to operate. Heat is removed from the core and transported to the reactor vessel wall by natural convection of primary sodium. Heat transport from the reactor vessel to the containment vessel is mainly by thermal radiation. The heat then thermally radiates to the air flowing around the containment vessel, which transports the heat to the atmosphere. Tests show that oxide layer formation and thermal cycling have negligible effect on RVACS performance. Analyses show that a sodium aerosol layer as thick as 1/4 inch on the air-side heat transfer surfaces will cause minimal loss of decay heat removal capability, permitting all core and vessel temperatures to remain well within limits under all DBE, BDBE, and all but two of the Deterministic Events. Furthermore, operation of RVACS with significant flow path blockage up to 90% is also acceptable. Even complete blockage of the air inlets is acceptable since two of the air outlets become inlets and permit air flow to resume.

Attribute No. 4 addresses additional safety systems to back up the two systems discussed above. PRISM has a Class 1E reactor protection system (RPS), six control rods, and two scram modes to back up the inherent negative reactivity feedback mechanism. The RPS is quad-redundant, fault-tolerant, fully automatic, and highly reliable. One control rod is capable of scramming the core at all times. Finally, there are two modes of scram, gravity and motor drive. This simple but reliable RPS, and two scram mode system, provide a highly reliable, redundant, and diverse shutdown safety system.

PRISM has one active and one passive decay heat removal system to back up the RVACS system. These are the auxiliary cooling system (ACS), and the normal condenser cooling system. The ACS is a high industrial grade heat removal system which is associated with the intermediate heat transfer system (IHTS) and the steam generator, and which provides for heat rejection from the steam generator to the atmosphere by means of natural air convection. The IHTS is designed to transport the decay heat from the reactor vessel to the steam generator by natural circulation of the intermediate sodium. Thus, the ACS is essentially a passive system, requiring only the opening of the air flow louvers to activate. The heat rejection capability of the ACS is a function of the average intermediate sodium temperature, and is sized to assure sufficient capability to prevent exceeding design limits in the event the normal heat sink is lost. The normal heat sink is the condenser cooling system, which, though an active system, requires only the pumping of condenser cooling water and feedwater to perform its function.

Attribute No. 5 concerning reliable, but non-safety grade BOP equipment is important to minimize the number of challenges to the safety systems. PRISM BOP equipment will be procured to the highest industrial grade standards. Augmenting this will be the learning curve improvement due to large numbers of units; the increased quality assurance achievable due to factory module fabrication, system assembly, checkout and control; and the ability to test the BOP equipment, along with the reactor, at plant startup and after each outage. Finally, the plant control system (PCS) is designed with a control rod runback feature which serves to limit both the number and severity of any transients which might be initiated, and to limit the number of challenges to the reactor protection system (RPS).

<u>Simplification (Systems/Analysis) - (Attribute No. 3)</u> - Simplicity of the PRISM design and analysis has been alluded to in several of the above discussions. For example, the small diameter reactor vessel confines the coolant flow path, and allows accurate analytical prediction of flow paths and thermal gradients. The safety systems are simplified because there are fewer of them; and those that are included rely either on inherent and passive mechanisms (negative reactivity feedback, RVACS, seismic isolation), or on a reduced number of active actions (RPS), which simplify analysis and operation.

Non-safety systems also benefit from the simplicity of the PRISM design. For example, the intermediate heat transfer system (IHTS) has a low operating temperature and pressure, utilizes a constant speed mechanical pump, and is designed to high quality industrial standards. These features allow large material design margins, a simplified control system, and ease of fabrication and system operation.

For another example, the steam generator system (SGS) utilizes a low pressure saturated steam cycle, and is designed to high quality industrial standards. The saturated cycle significantly simplifies the SGS control, eliminates the need for special startup equipment, and provides large thermal inertia to mitigate plant thermal transients. The low steam pressure also allows the IHX, IHTS, and SGS to be designed for full steam pressure, allowing inherent mitigation of a postulated multi-tube failure in the steam generator.

The final result of the simplicity of the PRISM concept is a power plant design that is straightforward to analyze, build, and operate. These features help ensure that normal and off-normal operation can be predicted with confidence, factored into the design of the plant and into the training of the operators, and demonstrated in full scale tests with the confidence that the results will confirm the predictions.

<u>Design Features That Are, Or Can Be, Proven (Attribute No. 9)</u> - Many of the design features incorporated into PRISM have already been proven in liquid metal testing programs performed in conjunction with the initial proof-in-principle tests (EBR-I, BR-5, SEFOR), the metal and oxide fuel test reac-

tors (DFR, EBR-II, RAPSODIE, BOR-60, KNK-2, JOYO, FFTF), the power demonstration plants (BN-350, PHENIX, PFR, SNR-300, MONJU), the commercial power plants (BN-600, SUPER PHENIX), the Clinch River Breeder Reactor Program, the current DOE R&D program, and other industrial experience.

Included in the above category are the inherent reactivity feedback mechanisms of the metal fuel, RVACS heat transfer characteristics, steam generator performance, in-vessel pantograph refueling machine design and operation, seismic isolator design and performance, flow characteristics inside the reactor vessel, self-cooled EM pump development, multi-module control design, and factory module design and fabrication. All of these features have been, or will be, demonstrated prior to use in PRISM. Thus, the PRISM design builds on the liquid metal reactor technology developed extensively in the United States and internationally, with a minimum of additional R&D required. The majority of the remaining R&D tasks are in the demonstration and qualification categories, not the new invention category. High risk innovations have not been included in the design.

In addition to the above, the major difference between PRISM and all other liquid metal reactors is the fact that for first time, a reactor has been designed that allows an affordable full scale safety test of design basis and key beyond design events. Such a safety test is planned for PRISM, using a fully prototypical reactor module and safety systems. This safety test will provide the final proof that the PRISM concept is safe and reliable, capable of meeting all safety requirements, with a degree of margin not heretofore achievable.