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MIRAGLIA,F. Licensing Branch 3

SEE REPORT & DRAWINGS

SUBJECT: Forwards info & drawings in response to NRC requirements identified in NUREG-0737 & NUREG-0600 items not addressed in NUREG-0737.Encl info will be incorporated into Amend 23 to FSAR scheduled for submittal on 810227.

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February 2, 1981

Director, Office of Nuclear Reactor Regulation
Attention: Mr. Frank Miraglia, Branch Chief
Licensing Branch No. 3
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

SCE's letters of August 16, 1979, and December 19, 1979, transmitted reports which discussed the status of the San Onofre Units 2 and 3 evaluation program relative to the TMI incident. The NRC issued NUREG-0660, NRC Action Plan Developed as a Result of the TMI-2 Accident, dated May, 1980 and by letter dated September 12, 1980, SCE transmitted Amendment No. 20 to the San Onofre Units 2 and 3 FSAR (a separate FSAR volume) directly addressing the NRC TMI requirements discussed in NUREG-0660.

Subsequently, the NRC issued NUREG-0737, Clarification of TMI Action Plan Requirements dated November, 1980, which identifies and clarifies the NUREG-0660 items which have been approved for implementation by the NRC as of November, 1980. Enclosed are sixty-three copies of SCE's responses to the NRC requirements identified in NUREG-0737. The enclosure also includes responses to NUREG-0660 items which were not addressed in NUREG-0737. The FSAR will be revised to reflect the enclosed information in Amendment No. 23 to the San Onofre Units 2 and 3 FSAR which is scheduled for submittal to the NRC on February 27, 1981.

If you have any questions or comments concerning this matter, please contact me.

NEED 2 Rids
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Very truly yours,

K P Baskin

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FOREWORD

This document is provided in response to the Nuclear Regulatory Commission (NRC) Action Plan NUREG 0660. This action plan imposes certain requirements upon applicants seeking NRC license to operate nuclear generating stations. Those requirements approved by the Commission for implementation are enumerated and, in some cases, further clarified in NUREG 0737. The data provided herein is in accordance with these requirements. 23

The information herein is sectionalized, each section consisting of a NUREG 0660/0737 requirement followed by the response of the applicant. Page, table, and figure numbers for each section consist of the NUREG requirement number followed by a - and sequential numbers beginning with the numeral 1.

A list of effective pages is included. This list provides an inventory of all text pages, tabs, and figures comprising this document.

CONTENTS

TABLES

FIGURES

I.A.1.1 - NUREG-0737 SHIFT TECHNICAL ADVISOR

REQUIREMENT

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Clarification

The letter of October 30, 1979 clarified the short-term STA requirements. That letter indicated that the STAs must have completed all training by January 1, 1981. This paper confirms these requirements and requests additional information. 23

The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue.

The staff has not yet established the detailed elements of the academic and training requirements of the STA beyond the guidance given in its October 30, 1979 letter. Nor has the staff made a decision on the level of upgrading required for licensed operating personnel and the man-machine interface in the control room that would be acceptable for eliminating the need of an STA. Until these requirements for eliminating the STA position have been established, the staff continues to require that, in addition to the staffing requirements specified in its July 31, 1980 letter (as revised by item I.A.1.3 of this enclosure), an STA be available for duty on each operating shift when a plant is being operated in Modes 1-4 for a PWR and Modes 1-3 for a BWR. At other times, an STA is not required to be on duty.

Since the October 30, 1979 letter was issued, several efforts have been made to establish, for the longer term, the minimum level of experience, education, and training for STAs. These efforts include work on the revision to ANS-3.1, work by the Institute of Nuclear Power Operations (INPO), and internal staff efforts.

INPO recently made available a document entitled "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training." A copy of Revision 0 of this document, dated April 30, 1980, is attached as Appendix C. Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STAs. The NRC staff finds that the descriptions as set forth in Sections 5 and 6 of Revision 0 to the INPO document are an acceptable approach for the selection and training of personnel to staff the STA positions. (Note: This should not be interpreted to mean that this is an NRC requirement at this time. The intent is to refer to the INPO document as acceptable for interim guidance for a utility in planning its STA program over the long term (i.e., beyond the January 1, 1981 requirement to have STAs in place in accordance with the qualification requirements specified in the staff's October 30, 1979 letter).)

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their STA training program and their plans for requalification training. This description shall indicate the level of training attained by STAs by January 1, 1981 and demonstrate conformance with the qualification and training requirements in the October 30, 1979 letter. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.

23 No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their long-term STA program, including qualification, selection criteria, training plans, and plans, if any, for the eventual phaseout of the STA program. (Note: The description shall include a comparison of the licensee/applicant program with the above-mentioned INPO document. This request solicits industry views to assist NRC in establishing long-term improvements in the STA program. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.)

RESPONSE

An on-site Shift Technical Advisor (STA) to the shift supervisor for San Onofre Units 2&3 will be provided for each shift, and will be available within 10 minutes of being summoned. The STA will serve both San Onofre Units 2&3. The duties of the STA will be promulgated by Station Order and will include:

- Diagnose accidents and off-normal events for their significance to reactor safety and advise the shift supervisor.
- Conduct ongoing review of the operating experience of San Onofre Units 2&3 and other plants of similar design from a reactor safety viewpoint.
- Interface with the Onsite Safety Engineering Group (see section I.B.1.2).

This Station Order will further emphasize the dedication to safety associated with the STA position and prohibit assigning the STA any duties associated with the commercial operation of the plant. Organizationally, the STA will report to the Supervisor, Nuclear Safety, who is independent from the operations staff.

Shift Technical Advisor (STA) training which meets the TMI lessons-learned requirements will be completed prior to fuel load.

A description of the current training program and the long-term STA program was previously transmitted to the NRC from Mr. J. G. Haynes to Mr. D. G. Eisenhut by letter dated December 30, 1980. 23

REFERENCE

Letter from Mr. J. G. Haynes to Mr. D. G. Eisenhut dated December 30, 1980.

I.A.1.2 NUREG 0660 SHIFT SUPERVISOR ADMINISTRATIVE DUTIES

REQUIREMENT (NUREG 0578, Item 2.2.1a)

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

RESPONSE

1. The Vice President of Nuclear Engineering and Operations will issue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties. This directive will be issued prior to receipt of the operating license and will be updated annually thereafter.

2. Plant procedures governing the shift supervisor and control room operators will be reviewed and revised as necessary to emphasize their duties, responsibility, and authority, including chain of command and relief procedures. The plant procedures will be implemented prior to receipt of the operating license.

3. The training program for shift supervisors will emphasize their responsibilities for safe operation and management of the plant. See sections I.A.2.3 (Administration of Training Programs) and I.A.3.1 (Revise Scope and Criteria for Licensing Examinations) for further discussions of the training program.

4. The administrative duties of the shift supervisor will be delineated in a station order which will be approved by the Manager of Nuclear Operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant will be delegated to other personnel. The station order will be implemented prior to fuel load.

REFERENCE

TMI items I.A.2.3 and I.A.3.1.

I.A.1.3 - NUREG 0737 SHIFT MANNING

REQUIREMENT

Position

This position defines shift manning requirements for normal operation. The letter of July 31, 1980 from D. G. Eisenhut to all power reactor licensees and applicants sets forth the interim criteria for shift staffing (to be effective pending general criteria that will be the subject of future rulemaking). Overtime restrictions were also included in the July 31, 1980 letter.

Clarification

Page 3 of the July 31, 1980 letter is superseded in its entirety by the following:

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation. 23

These administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, I&C technicians and key maintenance personnel).

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," dated February 1, 1980 discusses the concern of overtime work for members of the plant staff who perform safety-related functions.

The staff recognizes that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal 8-hour working day, the effects of shift rotation, and other factors. NRC has initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that of IE Circular No. 80-02.

In the event that overtime must be used (excluding extended periods of shutdown for refueling, major maintenance or major plant modifications), the following overtime restrictions should be followed.

1. An individual should not be permitted to work more than 12 hours straight (not including shift turnover time).

2. There should be a break of at least 12 hours (which can include shift turnover time) between all work periods.

3. An individual should not work more than 72 hours in any 7-day period.

4. An individual should not be required to work more than 14 consecutive days without having 2 consecutive days off.

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

If a reactor operator or senior reactor operator has been working more than 12 hours during periods of extended shutdown (e.g., at duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment.

23 NRC encourages the development of a staffing policy that would permit the licensed reactor operators and senior reactor operators to be periodically assigned to other duties away from the control board during their normal tours of duty.

If a reactor operator is required to work in excess of 8 continuous hours, he shall be periodically relieved of primary duties at the control board, such that periods of duty at the board do not exceed about 4 hours at a time.

The guidelines on overtime do not apply to the shift technical advisor provided he or she is provided sleeping accommodations and a 10-minute availability is assured.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning 90 days after July 31, 1980.

See section III.A.1.2 for minimum staffing and augment capabilities for emergencies.

RESPONSE

Station administrative procedures will be revised to reflect shift manning and overtime requirements and shall contain provisions governing the movement of key personnel, consistent with the guidelines of NUREG-0737.

REFERENCE

None.

I.A.2.1 - NUREG 0737 IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS

REQUIREMENT

Position

Effective December 1, 1980, an applicant for a senior reactor operator (SRO) license will be required to have been a licensed operator for 1 year.

Clarification

Applicants for SRO either come through the operations chain (C operator to B operator to A operator, etc.) or are degree-holding staff engineers who obtain licenses for backup purposes.

In the past, many individuals who came through the operator ranks were administered SRO examinations without first being an operator. This was clearly a poor practice and the letter of March 28, 1980 requires reactor operator experience for SRO applicants.

However, NRC does not wish to discourage staff engineers from becoming licensed SROs. This effort is encouraged because it forces engineers to broaden their knowledge about the plant and its operation. 23

In addition, in order to attract degree-holding engineers to consider the shift supervisor's job as part of their career development, NRC should provide an alternate path to holding an operator's license for 1 year.

The track followed by a high-school graduate (a non-degreed individual) to become an SRO would be 4 years as a control room operator, at least one of which would be as a licensed operator, and participation in an SRO training program that includes 3 months on shift as an extra person.

The track followed by a degree-holding engineer would be, at a minimum, 2 years of responsible nuclear power plant experience as a staff engineer, participation in an SRO training program equivalent to a cold applicant training program, and 3 months on shift as an extra person in training for an SRO position.

Holding these positions assures that individuals who will direct the licensed activities of licensed operators have had the necessary combination of education, training, and actual operating experience prior to assuming a supervisory role at that facility.

The staff realizes that the necessary knowledge and experience can be gained in a variety of ways. Consequently, credit for equivalent experience should be given to applicants for SRO licenses.

Applicants for SRO licenses at a facility may obtain their 1-year operating experience in a licensed capacity (operator or senior operator) at another nuclear power plant. In addition, actual operating experience in a position that is equivalent to a licensed operator or senior operator at military propulsion reactors will be acceptable on a one-for-one basis. Individual applicants must document this experience in their individual applications in sufficient detail so that the staff can make a finding regarding equivalency.

Applicants for SRO licenses who possess a degree in engineering or applicable sciences are deemed to meet the above requirement, provided they meet the requirements set forth in sections A.1.a and A.2 in enclosure 1 in the letter from H. R. Denton to all power reactor applicants and licensees, dated March 28, 1980, and have participated in a training program equivalent to that of a cold senior operator applicant.

23 NRC has not imposed the 1-year experience requirement on cold applicants for SRO licenses. Cold applicants are to work on a facility not yet in operation; their training programs are designed to supply the equivalent of the experience not available to them.

RESPONSE

Applicants for Senior Reactor Operator (SRO) license will have qualifications consistent with the guidelines of NUREG-0737.

REFERENCE

None.

I.A.2.3 - NUREG 0737 ADMINISTRATION OF TRAINING PROGRAMS

REQUIREMENT

Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

Clarification

The above position is a short-term position. In the future, accreditation of training institutions will include review of the procedure for certification of instructors. The certification of instructors may, or may not, include successful completion of an SRO examination.

The purpose of the examination is to provide NRC with reasonable assurance during the interim period, that instructors are technically competent. 23

The requirement is directed to permanent members of training staff who teach the subjects listed above, including members of other organizations who routinely conduct training at the facility. There is no intention to require guest lecturers who are experts in particular subjects (reactor theory, instrumentation, thermodynamics, health physics, chemistry, etc.) to successfully complete an SRO examination. Nor is it intended to require a system expert, such as the instrument and control supervisor teaching the control rod drive system, to sit for an SRO examination.

RESPONSE

Permanent members of the station training staff who teach the subjects identified in NUREG-0737 will demonstrate senior reactor operator (SRO) qualifications. Instructors who are not SRO licensed will participate in the SRO exams for San Onofre Unit 2. Instructors will also participate in appropriate requalification programs.

REFERENCES

None.

I.A.3.1 - NUREG 0737 <u>REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS--SIMULATOR EXAMS (ITEM 3)</u>

REQUIREMENT

Position

Simulator examinations will be included as part of the licensing examinations.

Clarification

The clarification does not alter the staff's position regarding simulator examinations.

The clarification does provide additional preparation time for utility companies and NRC to meet examination requirements as stated. A study is under way to consider how similar a nonidentical simulator should be for a valid examination. In addition, present simulators are fully booked months in advance.

Application of this requirement was stated on June 1, 1980 to applicants where a simulator is located at the facility. Starting October 1, 1981, simulator examinations will be conducted for applicants of facilities that do not have simulators at the site. 23

NRC simulator examinations normally require 2 to 3 hours. Normally, two applicants are examined during this time period by two examiners.

Utility companies should make the necessary arrangements with an appropriate simulator training center to provide time for these examinations. Preferably these examinations should be scheduled consecutively with the balance of the examination. However, they may be scheduled no sooner than 2 weeks prior to and no later than 2 weeks after the balance of the examination.

RESPONSE

Simulator training will be provided to license candidates; this training will include sufficient preparation for simulator examinations, as required by the NRC.

REFERENCE

None.

I.B.1.2 - NUREG 0737 INDEPENDENT SAFETY ENGINEERING GROUP

REQUIREMENT

Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities including maintenance, modifications, operational problems, operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization. 23

Clarification

The new ISEG shall not replace the plant operations review committee (PORC) and the utility's independent review and audit group as specified by current staff guidelines (Standard Review Plan, Regulatory Guide 1.33, Standard Technical Specifications). Rather, it is an additional independent group of a minimum of five dedicated, full-time engineers, located onsite, but reporting offsite to a corporate official who holds a high-level, technically oriented position that is not in the management chain for power production. The ISEG will increase the available technical expertise located onsite and will provide continuing, systematic, and independent assessment of plant activities. Integrating the shift technical advisors (STAs) into the ISEG in some way would be desirable in that it could enhance the group's contact with and knowledge of day-to-day plant operations and provide additional expertise. However, the STA on shift is necessarily a member of the operating staff and cannot be independent of it.

It is expected that the ISEG may interface with the quality assurance (QA) organization, but preferably should not be an integral part of the QA organization.

The functions of the ISEG require daily contact with the operating personnel and continued access to plant facilities and records. The ISEG review functions can, therefore, best be carried out by a group physically located onsite. However, for utilities with multiple sites, it may be possible to perform portions of the independent safety assessment function in a centralized location for all the utility's plants. In such cases, an onsite group still is required, but it may be slightly smaller than would be the case if it were performing the entire independent safety assessment function. Such cases will be reviewed on a case-by-case basis.

At this time, the requirement for establishing an ISEG is being applied only to applicants for operating licenses in accordance with Action Plan item I.B.1.2. The staff intends to review this activity in about a year to determine its effectiveness and to see whether changes are required. Applicability to operating plants will be considered in implementing long-term improvements in organization and management for operating plants (Action Plan item I.B.1.1).

23

RESPONSE

Figure I.B.1.2-1 illustrates the composition of the ISEG and its relationship with other groups within the Nuclear Engineering and Operations Department and with the Quality Assurance Organization.

Shift Technical Advisors (STAs) will stand watch on a duty day basis and thus will be asleep at times, but will be available in the Control Room on short notice. The STA's primary responsibility is to provide technical assistance to the shift supervisor during an off-normal event. When on shift but not assisting the shift supervisor, and when off-duty, STAs will perform the functions of the ISEG. Functions of the ISEG are listed below:

1. Evaluation of operating experience information.

   Operating experience reports from INPO/NSAC, and other appropriate sources of information are screened to determine applicability to San Onofre by the Nuclear Safety Group. Applicable information is forwarded to the ISEG for a detailed review and development of recommendations for procedural revision design changes, and other areas which improve plant safety.

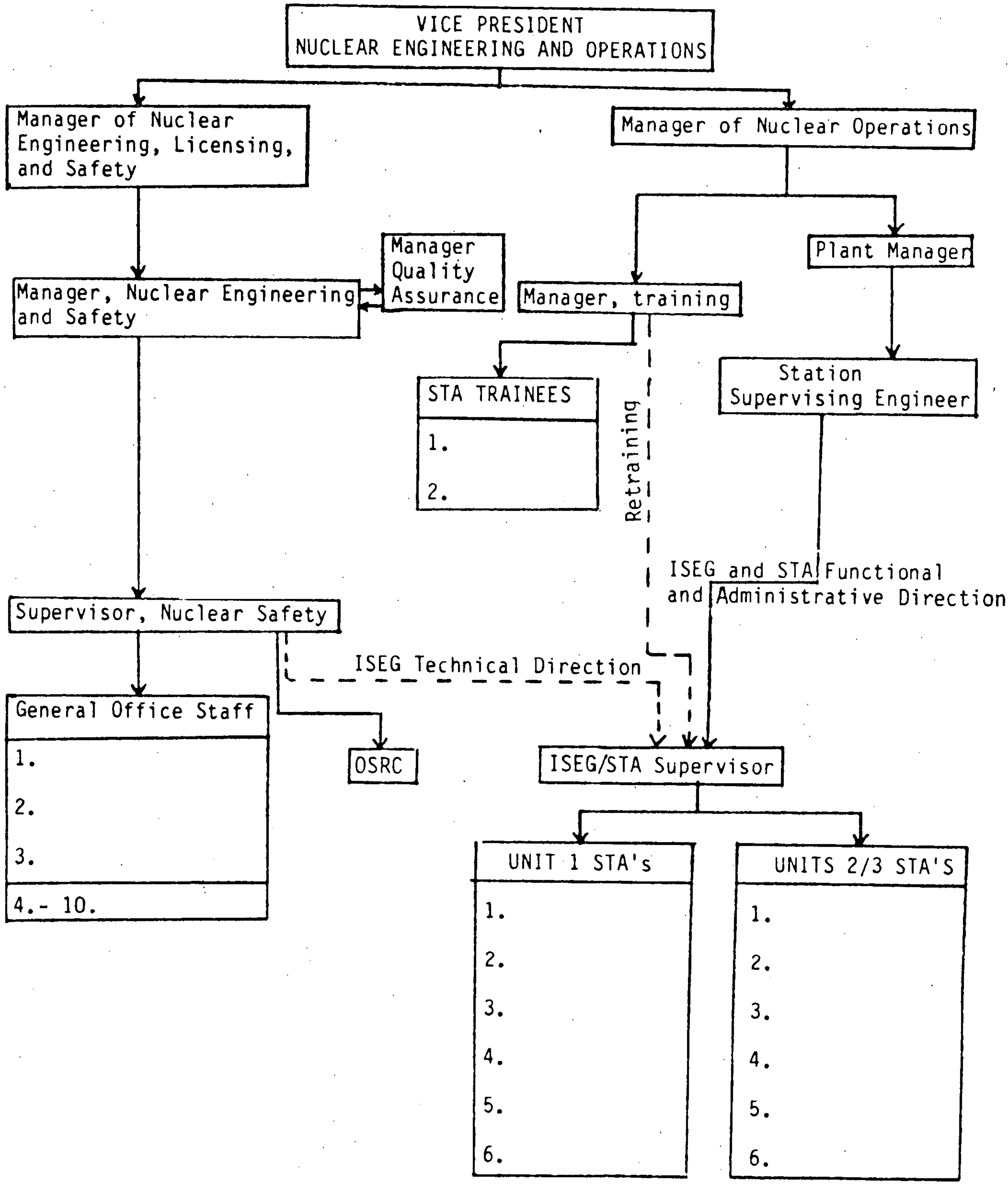
2. Independent evaluation and surveillance of plant activities.

   A wide range of plant activities including operations, maintenance, and modifications are monitored during the actual performance of the work to evaluate the technical adequacy of the methods used, recommend equipment changes, and aid in the establishment of programmatic requirements for plant activities. This surveillance is also intended to provide independent verification that these activities are performed correctly and ensure that the potential for human error is reduced as far as possible.

The Nuclear Safety Group is a staff organization consisting of a minimum of a supervisor and three staff specialists who provide company independent reviews and audits. The ISEG receives technical direction from the Nuclear Safety Group and reports administratively and functionally to the Station Supervising Engineer. The ISEG does not replace either the company review committee or the plant onsite review committee (OSRC). 23

REFERENCE

None



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

ISEG Organization

Figure I.B.1.2-1

I.C.1 - NUREG 0737 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

REQUIREMENT

Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (also refer to item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow-up on the bulletin and orders matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C (refer to Table C.1, 23
items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, items 4, 12, 17, 18, 19, 20; and Table C.3, items 6, 35, 37, 38, 39, 41, 47, 55, 57).

Clarification

The letters of September 13 and 27, October 10 and 30, November 9, 1979, required that procedures and operator training be developed for transients and accidents. The initiating events to be considered should include the events presented in the final safety analysis report (FSAR) loss of instrumentation buses, and natural phenomena such as earthquakes, floods, and tornadoes. The purpose of this paper is to clarify the requirements and add additional requirements for the reanalysis of transients and accidents and inadequate core cooling.

Based on staff reviews to date, there appear to be some recurring deficiencies in the guidelines being developed. Specifically, the staff has found a lack of justification for the approach used (i.e., symptom-, event-, or function-oriented) in developing diagnostic guidance for the operator and in procedural development. It has also been found that although the guidelines take implicit credit for operation of many systems or components, they do not address the availability of these systems under expected plant conditions nor do they address corrective or alternative actions that should be performed to mitigate the event should these systems or components fail.

The analyses conducted to date for guideline and procedure development contain insufficient information to assess the extent to which multiple failures are considered. NUREG-0578 concluded that the single-failure criterion was not considered appropriate for guideline development and called for the consideration of multiple failures and operator errors. Therefore, the analyses that support guideline and procedure development should consider the occurrences of multiple and consequential failures. In general, the sequence of events for the transients and accidents and inadequate core cooling analyzed should postulate multiple failures such that, if the failures were unmitigated, conditions of inadequate core cooling would result.

Examples of multiple failure events include:

1. Multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator.

2. Failure of main and auxiliary feedwater.

3. Failure of high-pressure reactor coolant makeup system.

4. An anticipated transient without scram (ATWS) event following a loss of offsite power, stuck-open relief valve or safety/relief valve, or loss of main feedwater.

23

5. Operator errors of omission or commission.

The analyses should be carried out far enough into the event to assure that all relevant thermal/hydraulic/neutronic phenomena are identified (e.g., upper head voiding due to rapid cooldown, steam generator stratification). Failures and operator errors during the long-term cooldown period should also be addressed.

The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling procedure including the use of instrumentation to identify inadequate core cooling conditions. Rationale for this transition should be discussed. Additional information that should be submitted includes:

1. A detailed description of the methodology used to develop the guidelines;

2. Associated control function diagrams, sequence-of-event diagrams, or others, if used;

3. The bases for multiple and consequential failure considerations;

4. Supporting analysis, including a description of any computer codes used; and

5. A description of the applicability of any generic results to plant-specific applications.

Owners' group or vendor submittals may be referenced as appropriate to support this reanalysis. If owners' group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

Pending staff approval of the revised analysis and guidelines; the staff will continue the pilot monitoring of emergency procedures described in task action plan item I.C.8 (NUREG-0660). For PWRs, this will involve review of the loss of coolant, steam-generator-tube rupture, loss of main feedwater, and inadequate core cooling procedures. The adequacy of each PWR vendor's guidelines will be identified to each NTOL during the emergency-procedure review. Since the analysis and guidelines submitted by the General Electric Company (GE) owners' group that comply with the requirements stated above have been reviewed and approved for trial implementation on six plants with applications for operating licenses pending, the interim program for BWRs will consist of trail implementation on these six plants.

Following approval of analysis and guidelines and the pilot monitoring of emergency procedures, the staff will advise all licensees of the adequacy of the guidelines for application to their plants. Consideration will be given to human factors engineering and system operational characteristics, such as information transfer under stress, compatibility with operator training and control-room design, the time required for component and system 23
response, clarity of procedural actions, and control-room-personnel interactions. When this determination has been made by the staff, a long-term plan for emergency procedure review, as described in task action plan item I.C.9, will be made available. At that time, the reviews currently being conducted on NTOLs under item I.C.8 will be discontinued, and the review required for applicants for operating licenses will be as described in the long-term plan. Depending on the information submitted to support development of emergency procedures for each reactor type or vendor, this transition may take place at different times. For example, if the GE guidelines are shown to be effective on the six plants chosen for pilot monitoring, the long-term plan for BWRs may be complete in early 1981. Operating plants and applicants will then have the option of implementing the long-term plan in a manner consistent with their operating schedule, provided they meet the final date required for implementation. This may require a plant that was reviewed for an operating license under item I.C.8 to revise its emergency procedures again prior to the final implementation date for Item I.C.9. The extent to which the long-term program will include review and approval of plant-specific procedures for operating plants has not been established. Our objective, however, is to minimize the amount of plant-specific procedure review and approval required. The staff believes this objective can be acceptably accomplished by concentrating the staff review and approval on generic guidelines. A key element in meeting this objective is the use of staff-approved generic guidelines and guideline revisions by licensees to develop procedures. For this approach to be effective, it is imperative that, once the staff has issued approval of a guideline, subsequent revisions of the guideline should not be implemented

by licensees until reviewed and approved by the staff. Any changes in plant-specific procedures based on unapproved guidelines could constitute an unreviewed safety issue under 10 CFR 50.59. Deviations from this approach on a plant-specific basis would be acceptable provided the basis is submitted by the licensee for staff review and approval. In this case, deviations from generic guidelines should not be implemented until staff approval is formally received in writing. Interim implementation of analysis and procedures for small-break loss-of-coolant accident and inadequate core cooling should remain on the schedule contained in NUREG-0578, Recommendation 2.1.9.

RESPONSE

Southern California Edison (SCE) has participated in C-E Owners Group activities conducted since the Three Mile Island accident to develop improved emergency procedure guidelines and associated supporting analyses. The C-E Owners Group has completed numerous documents which have ben submitted to the NRC for review. The C-E Owners Group is currently sponsoring further activities which will be completed in the first half of 1981 and documented at that time. A summary of results obtained to date and current activities is approved below.

The initial C-E Owners Group analysis of Inadequate Core Cooling (ICC) is
documented in report CEN-117, "Inadequate Core Cooling - A Response to NRC
IE Bulletin 79-06C, Item 6 for Combustion Engineering Nuclear Steam Supply
23 Systems". This report was submitted to the NRC staff for review on
October 31, 1979, by the C-E Owners Group. SCE referred to this report in
its letter dated December 12, 1979 which was submitted in response to documentation requirements of the Short-Term Lessons Learned.

"Operational Guidance for Inadequate Core Cooling" was prepared by the C-E Owners' Group based on the analyses in report CEN-117. This operational guidance was distributed to all members of the C-E Owners Group for their use in review and possible revision of plant emergency procedures in December, 1979. A copy of this operational guidance was submitted to the NRC staff for review by the C-E Owners' Group on December 10, 1980.

Since early 1980, the C-E Owners Group has sponsored an extensive study of instrumentation response characteristics under ICC conditions. This study was described to the NRC staff at a meeting in Bethesda, MD, on May 28, 1980. This study was completed in December, 1980, and its results have been distributed to members of the C-E Owners Group for their use. SCE is currently evaluating the results of this study for use in possible revisions to plant emergency procedures. Such revisions would be based upon determination of the usefulness of specific instrumentation for detection of ICC. This evaluation and subsequent revision of plant emergency procedures as required is expected to be completed by the required date of the first refueling outage after January 1, 1982.

The initial C-E Owners Group analyses of transients and accidents (non-LOCA) is documented in report CEN-128, "Response of Combustion Engineering Nuclear

Steam Supply System to Transients and Accidents". This report was submitted to the NRC staff for review on April 1, 1980. The results in this report show how a typical C-E-designed plant would most likely respond to various event initiators and shows what systems are actuated following each event. The report includes results of plant simulation analyses with digital computer codes to determine transient behavior of pertinent plant process parameters, components, and systems and results of sequence of events analyses performed to identify component and system functions and alternate means to accomplish specified safety functions.

The analyses contained in report CEN-128 consider single active failure for each system called upon to function for a particular event. Passive failures and multiple system failures are not considered. The sequence of events analyses (SEA) show the various paths through an event without probabilistic considerations. Each SEA demonstrates how specified safety functions are satisfied. Sequence of events diagrams (SED) are used to show how these functions are accomplished and include single active failures in each responding system and operator failure to perform manual actions. Consequential failures are considered in the SED for the steam line break.

Since early 1980, the C-E Owners Group has conducted a program to develop analyses of transients and accidents involving multiple failures. These analyses were outlined to the NRC staff in a meeting held in Bethesda, MD, on January 31, 1980. These analyses are currently scheduled to be completed in the first quarter of 1981. The results of these analyses will provide one basis for possible revision of emergency procedure guidelines. 23

The initial C-E Owners Group development of emergency procedure guidelines was completed in the first quarter of 1980. These emergency procedure guidelines are documented in report CEN-128. This report was submitted to the NRC staff for review on April 1, 1980. CEN-128 was summarized and referenced in the San Onofre Units 2 and 3 response to NUREG-0660 provided with Amendment 20 to the FSAR.

The emergency procedure guidelines contained in report CEN-128 were prepared based on extensive reviews of existing emergency procedures, past safety, and design analyses, the plant simulation and sequence of events analyses in CEN-128, and interviews with operations personnel at plants with operating C-E reactors. These emergency procedure guidelines were prepared to be used as a basis for reviewing, and revising if necessary, existing plant emergency procedures. SCE has reviewed these emergency procedure guidelines and has made revisions to its plant emergency procedures as necessary.

The NRC staff, in a letter dated July 17, 1980, sent questions to the C-E Owners Group concerning the emergency procedure guidelines documented in report CEN-128. A meeting was held with the NRC staff in Bethesda, MD, on September 11, 1980, to discuss these questions and answers to them. The C-E Owners Group is presently preparing answers to these questions and revisions to the emergency procedure guidelines in report CEN-128 as are appropriate. A preliminary response to these questions was submitted by the C-E Owners Group to the NRC staff in a letter dated December 10, 1980. The remaining responses will be submitted to the NRC staff by the C-E Owners Group in mid-January, 1981.

Since early 1980, the C-E Owners Group has conducted an extensive evaluation of specific technical characteristics of emergency procedure guidelines. These include (1) the diagnostic guidance to be provided in emergency procedure guidelines, (2) the need for a separate guideline for inadequate core cooling, and (3) the format for presentation of emergency guidance. This evaluation is currently scheduled to be completed in the first quarter of 1981. The results of this evaluation will serve as one basis for possible revision of emergency procedure guidelines contained in report CEN-128.

The C-E Owners Group agreed on December 3, 1980, to conduct a series of workshops concerning emergency procedure guidelines in early 1981. These workshops were intended to provide a formal process by which the emergency procedure guidelines documented in report CEN-128 will be revised to account for multiple failure considerations. Input to these workshops will be provided by the analysis and emergency procedure guidelines studies which have been conducted by the C-E Owners Group since early 1980. The workshops are to be attended by staff personnel from C-E and from utilities which own C-E reactors. These workshops will also provide the opportunity to explore multiple-failure scenarios beyond those which have been currently identified in the C-E Owners Group analyses of transients and accidents. LOCA will also be considered in these workshops.

23 The C-E Owners Group has initiated an effort to define the process by which plant emergency procedures should be developed or modified using emergency procedure guidelines and supporting analyses. SCE is participating in this definition process and feels that its completion is necessary before plant emergency procedures are further revised based on revisions to the emergency procedure guidelines. This C-E Owners Group activity is scheduled to be completed by May 1, 1981.

Following the completion of the studies currently being conducted by the C-E Owners Group and the emergency procedure guidelines workshops, a revised set of emergency procedure guidelines will be submitted for review to the NRC staff by the C-E Owners Group. The C-E Owners Group has scheduled a meeting with the NRC staff for January, 1981 in order to discuss the process being used for revision of emergency procedure guidelines. The revised emergency procedure guidelines are currently scheduled to be submitted for review to the NRC staff by the C-E Owners Group on June 1, 1981.

Following completion of the NRC review of the revised emergency procedure guidelines and completion of the C-E Owners Group activities to define the process for development or revision of plant emergency procedures, SCE will evaluate the need for revision of its plant emergency procedures. The schedule in NUREG-0737 indicates that six months will be required for NRC staff review and approval and that another six months or more are to be allowed for revision and implementation of emergency procedures. Therefore, the San Onofre Units 2 and 3 emergency procedures will be revised if necessary after December 1, 1981, and the revisions implemented at the first refueling outage after June 1, 1982.

I.C.2 - NUREG 0660 SHIFT AND RELIEF TURNOVER PROCEDURES

REQUIREMENT (NUREG 0578, Item 2.2.1.c)

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist.

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist).

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist).

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

RESPONSE

The applicants are reviewing and revising plant procedures related to shift and relief turnover procedures. These revised procedures will include checklists that will assure a) critical plant parameters are within allowable limits, b) availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents, and c) identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. The existing Engineering Safety Features Inoperable/Bypass Status Display will aid in performance of the above responsibilities.

Shift turnover procedures will be developed that identify equipment under maintenance or test that by themselves could degrade safety systems.

The above procedures will be periodically audited to ensure that they are effectively utilized.

REFERENCE

None

I.C.3 - NUREG 0660 SHIFT SUPERVISOR RESPONSIBILITIES

This subject is discussed in section I.A.1.2, Shift Supervisor Administrative Duties.

I.C.4 - NUREG 0660 CONTROL ROOM ACCESS

REQUIREMENT

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access, and

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

RESPONSE

1. The applicants have developed an administrative station order governing control room access. The order assigns the authority and responsibility for limiting control room access to the watch engineer (shift supervisor).

2. In the event of an emergency, a clear line of authority and responsibility in the control room has been established. The line of succession is limited to persons possessing a current senior reactor operator's license. The Units 2 and 3 Emergency Plan identifies lines of communication and authority for plant management personnel 23

REFERENCE

None

I.C.5 - NUREG 0737 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

REQUIREMENT

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

1. Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

2. Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders); 23

3. Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

4. Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

5. Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

6. Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

7. Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Clarification

Each utility shall carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. In connection with this assessment function, it is important that procedures exist to assure that important information on operating experience originating both within and outside the organization is continually provided to operators and other personnel and that it is incorporated into plant operating procedures, training, and retraining programs.

Those involved in the assessment of operating experience will review information from a variety of sources. These include operating information from the licensee's own plant(s), publications such as IE Bulletins, Circulars, Notices, and pertinent NRC or industrial assessments of operating experience. In some cases, information may be of sufficient imporatnce that it must be dealt with promptly (through instructions, changes to operating and emergency procedures, issuance of special changes to operating and emergency procedures, issuance of special precautions, etc.) and must be handled in such a manner to assure that operations management personnel would be directly involved in the process. In many other cases, however, important information will become available which should be brought to the attention of
23 operators and other personnel for their general information to assure continued safe plant operation. Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high-priority matters are dealt with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the detriment of their overall proficiency. It is important, also, that technical reviews be conducted to preclude premature dissemination of conflicting or contradictory information.

RESPONSE

As indicated in our response to NUREG-0737 position I.B.1.2, a nuclear safety group has been established that will be responsible for establishing a program that provides for feedback of both industry and station operating experience. The INPO/NSAC Significant Event Evaluation and Information Network (SEE-IN) program will be utilized to screen and assess all operating experience information originating from outside of SCE.

The following information addresses **positions 1. through 7. above:**

1. Procedures will be developed which identify organizational responsibilities for review of operating experience originating from within (San Onofre LERs) and outside (SEE-IN results) the SCE organization, including feedback of pertinent information to station personnel and incorporation of such information into training and retraining programs.

2. The procedures will identify the steps necessary to review the above operating experience information and forward recommendations to the appropriate organization for implementing action.

3. The procedures will include means through which information requiring further dissemination within the station organization can be readily related to the job function of the recipients.

4., 5., & 6. These requirements are satisfied by the operation of the SEE-IN program. SCE procedures will identify how information from INPO is disseminated to station personnel. 23

7. INPO/NSAC technical activities are subject to the scrutiny of industry review groups. Thus, a separate audit of SEE-IN program activities by SCE is unnecessary. SCE will perform a periodic internal audit to determine the effectiveness of the feedback.

REFERENCE

Response to NUREG 0737 item I.B.1.2.

I.C.6 NUREC 0737 GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

REQUIREMENT

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3.

Clarification

Item I.C.6 of the U.S. Nuclear Regulatory Commission Task Action Plan (NUREG-0660) and Recommendation 5 of NUREG-0585 propose requiring that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided. An acceptable program for verification of operating activities is described below.

The American Nuclear Society had prepared a draft revision to ANSI Standard N18.7-1972 (ANS 3.2) "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." A second proposed revision to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)", which is to be issued for public comment in the near future, will endorse the latest draft revision to ANS 3.2 subject to the following supplemental provisions:

(1) Applicability of the guidance of Section 5.2.6 should be extended to cover surveillance testing in addition to maintenance.

(2) In lieu of any designated senior reactor operator (SRO), the authority to release systems and equipment for maintenance or surveillance testing or return-to-service may be delegated to an on-shift SRO, provided provisions are made to ensure that the shift supervisor is kept fully informed of system status.

(3) Except in cases of significant radiation exposure, a second qualified person should verify correct implementation of equipment control measures such as tagging of equipment.

(4) Equipment control procedures should include assurance that control-room operators are informed of changes in equipment status and the effects of such changes.

(5) For the return-to-service of equipment important to safety, a second qualified operator should verify proper systems alignment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are correctly aligned.

Note: A licensed operator possessing knowledge of the systems involved and the relationship of the systems to plant safety would be a "qualified" person. The staff is investigating the level of qualification necessary for other operators ot perform these functions.

For plants that have or will have automatic system status monitoring as discussed in Task Action Plan item I.D.3, NUREG-0660, the extent of human verification of operations and maintenance activities will be reduced. However, the need for such verification will not be eliminated in all instances.

RESPONSE

Provisions for human verification of operating and maintenance activities will be addressed within station procedures. The extent to which human verification will be addressed within station procedures will depend upon the amount of automatic system status monitoring equipment installed.

The items which are identified as supplemental provisions to ANS 3.2 will also be addressed within station procedures, to the extent that they are applicable to San Onofre Units 2 and 3.

I.C.7 - NUREG 0660 NSSS VENDOR REVIEW OF PROCEDURES

REQUIREMENT

Operating license applicants are required to obtain reactor vendor review of their low-power, power ascension and emergency procedures as a further verification of the adequacy of the procedures.

RESPONSE

The low-power and power ascension test and emergency procedures for San Onofre Units 2&3 are in the process of preparation and review. The NSSS vendor, Combustion Engineering (C-E), Inc., is preparing the low-power physics and power ascension test procedures for use by SCE during the startup of San Onofre Units 2&3. The C-E Site Manager is a member of the Test Working Group and participates in the review and approval of the low-power physics and power-ascension test procedures. In addition, C-E will be reviewing the specific emergency procedures listed in table I.C.7-1. Documentation will be available prior to the start of low-power testing which will verify that the NSSS vendor reviewed and approved procedures involved with the following:

- Preoperational and acceptance tests (FSAR table 14.2-3)
- Precritical tests (FSAR table 14.2-4)
- Low-power physics tests (FSAR table 14.2-5)
- Power ascension tests (FSAR table 14.2-6)
- Emergency Procedures (table I.C.7-1)

Amplifying information for the Units 2&3 startup test program can be found in FSAR section 14.2, which also lists the tests involved and the test organization.

REFERENCES

FSAR paragraph 13.1.1.1.2.3 has been revised. 21

Table I.C.7-1
EMERGENCY PROCEDURES

Title
Emergency Action Guidelines
Reactor Trip
Loss of Load
Loss of AC Power
Loss of Reactor Coolant Flow
Loss of Coolant Accident
Reactor Coolant Leak
CEDM Failure
Steam Line Rupture
Emergency Boration
Inadvertent Containment Isolation
Loss of Shutdown Cooling
Loss of Flux Indication/Flux Anomaly
High Activity in Reactor Coolant or Off-Gas
Emergency Plant Shutdown
Reactor Regulating System Malfunction
Pressure Control System Malfunction
Shutdown of Plant From Outside Control Room
Loss of Boron/Excessive Dilution
Loss of Reactor Coolant Makeup
Loss of Reactor Coolant Letdown
Loss of Reactor Coolant Purification
Damage of Spent Fuel
Loss of Pressurizer Level Control
Degraded Electrical Power
Reactor Coolant Pump/Motor Emergency

I.C.8 - NUREG-0660 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR-TERM OPERATING LICENSE APPLICANTS

REQUIREMENT (NUREG 0694)

The NRC will conduct an interdisciplinary and interoffice audit of selected plant emergency operating procedures (e.g. small-break LOCA, loss of feed-water, restart of engineered safety features following a loss of ac power, steam-line break, or steam-generator tube rupture).

The licensee should correct, before full-power operation, any deficiencies in the emergency procedures, as necessary, based on the NRC audit.

RESPONSE

Plant emergency procedures will be completed and available for NRC review and audit 6 months prior to fuel loading.

Any emergency procedure deficiencies identified during the NRC review and audit will be corrected and implemented prior to full power operation.

As indicated in the response to NRC NUREG-0737 Position I.C.1, SCE is currently involved with the NRC PTRB in the detailed review and revision of the following four (4) Emergency Operating Instructions (EOI's):

- Loss of Coolant Accident 23
- Steam Generator Tube Rupture
- Loss of Feedwater
- Determination of Adequate Core Cooling

REFERENCE

Response to NRC Position I.C.1.

I.D.1 - NUREG 0737 CONTROL-ROOM DESIGN REVIEWS

REQUIREMENT

Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control-room design review to identify and correct design deficiencies. This detailed control-room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

Clarification

NRR is presently developing human engineering guidelines to assist each licensee and applicant in performing detailed control-room review. A draft of the guidelines has been published for public comment as NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation." The due date for comments on this draft document was September 29, 1980. NRR will issue the final version of the guidelines as NUREG-0700, by February 1981, after receiving, reviewing, and incorporating substantive public comments from operating reactor licensees, applicants for operating licenses, human factors engineering experts, and other interested parties. NRR will issue evaluation criteria, by July 1981, which will be used to judge the acceptability of the detailed reviews performed and the design modifications implemented. 23

Applicants for operating licenses who will be unable to complete the detailed control-room design review prior to issuance of a license are required to perform a preliminary control-room design assessment to identify significant human factors problems. Applicants will find it of value to refer to the draft document NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation" in performing the preliminary assessment. NRR will evaluate the applicants' preliminary assessments including the performance by NRR of onsite review/audit. The NRR onsite review/audit will be on a schedule consistent with licensing needs and will emphasize the following aspects of the control room:

A. The adequacy of information presented to the operator to reflect plant status for normal operation, anticipated operational occurrences, and accident conditions;

B. The groupings of displays and the layout of panels;

C. Improvements in the safety monitoring and human factors enhancement of controls and control displays;

D. The communications from the control room to points outside the control room, such as the onsite technical support center, remote shutdown panel, offsite telephone lines, and to other areas within the plant for normal and emergency operation.

E. The use of direct rather than derived signals for the presentation of process and safety information to the operator;

F. The operability of the plant from the control room with multiple failures of nonsafety-grade and nonseismic systems;

G. The adequacy of operating procedures and operator training with respect to limitations of instrumentation displays in the control room;

H. The categorization of alarms, with unique definition of safety alarms.

I. The physical location of the shift supervisor's officer either adjacent
23 to or within the control-room complex.

Prior to the onsite review/audit, NRR will require a copy of the applicant's preliminary assessment and additional information which will be used in formulating the details of the onsite review/audit.

RESPONSE

Southern California Edison Company (SCE) established a Control Room Design Review (CRDR) Task Force and performed the preliminary assessment of the SONGS 2&3 Control Room. The Task Force was comprised of SCE Engineering, SCE Operations, Bechtel Power Corporation (BPC) Engineering, Combustion Engineering (CE), NSSS Engineering, and the SCE Consultant-Whitston Associates Human Factors Engineering personnel.

The Task Force considered Human Factor enhancement throughout the CRDR evaluation via the SCE Consultant and utilization of the draft NUREG/CR-1580 "Human Engineering Guide to Control Room Evaluation."

SCE also analyzed the NRC Control Room Audit Reports of other Utility Facilities to establish the SCE guidelines established prior to conducting the CRDR. These guidelines were reevaluated after the NRC Control Room audit of the SONGS 2 Control Room.

The CRDR was performed in two phases. The first phase addressed those items considered to have the highest impact on safe operation, normal and emergency, and enhancing the operator capability to make accurate decisions and take fast accurate actions. The following areas were reviewed.

- System Demarcation
- System Component Grouping
- Panel Labeling Hierarchy and Label Terminology
- Alarm Prioritization
- Instrument Scales and Coding
- Emergency Operating Instruction
- Environmental, Lighting and Acoustics

The review resulted in SCE initiating implementation of the following:

- Color Demarcation
- New Labeling Criteria (Hierarchy, Terminology and Top Orientation)
- Instrument Relocation (approx. 150)
- Applicable Scale Coding (operating range and alarm) 23
- Four Level Alarm Prioritization
- Interim Reduction Instrument Glare

The second phase addressed items which were reviewed as having impact of a lesser degree on the operator actions.

The following areas were included and reviewed in this category.

- Annunciator Alarm System vs. Plant Computer Alarms
- Remote Shutdown Panel
- Environmental-Lighting
- Environmental-Acoustics
- Communications
- Reevaluation of Instrument Glare
- Instrument Suitability

SCE is in the process of finalizing the results of the above reviews and will be initiating implementation direction consistent with the implementation schedule of NUREG-0700.

The results of the Task Force review are set forth in a two phase report, consistent with the above, which covers the objectives, review techniques, potential and real problem areas, criteria for solution of the problems and recommendations.

The NRC Audit of the San Onofre Units 2 and 3 control room was conducted
23 the week of August 4-8, 1980, including a walkdown of selected Emergency Instructions. The NRC Audit comments have been reviewed and the SCE position established. These were reviewed with the NRC in a meeting on September 16, 1980 and the mutually accepted Audit report was transmitted from SCE to the NRC by letter dated October 29, 1980.

SCE has completed the initial CRDR review implementation of the appropriate recommendations will be implemented in accordance with the above Audit report and NUREG-0700. On receipt of NUREG-0700, SCE will review the existing SCE CRDR report versus the NUREG-0700 document and establish a plan for implementation.

REFERENCE

None

I.D.2 - NUREG 0737 PLANT SAFETY PARAMETER DISPLAY CONSOLE

REQUIREMENT

Position

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console, each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

23

Clarification

These requirements for the SPDS are being developed in NUREG-0696, which is scheduled for issuance in November 1980.

RESPONSE

Southern California Edison will respond to the guidance provided in NUREG-0696 when the document has been received and reviewed. The Critical Functions Monitoring System being installed in San Onofre Units 2 and 3 is discussed in the response to TMI item III.A.1.2 concerning emergency support facilities and in FSAR sections 7.1.1.6, 7.1.1.7, 7.6.1.3, and 7.6.2.3.

REFERENCES

FSAR Sections 7.1.1.6, 7.1.1.7, 7.6.1.3 and 7.6.2.3; TMI item III.A.1.2.

I.G.1 - NUREG 0660 TRAINING DURING LOW-POWER TESTING

REQUIREMENT

NRR will require new operating licensees to conduct a set of low-power tests. The set of tests will be determined on a case-by-case basis for the first few plants. Then NRR will develop acceptance criteria for low-power test programs to provide "hands on" training for plant evaluation and off-normal events for each operating shift. It is not expected that all tests will be required to be conducted by each operating shift. Observation by one shift of training of another shift may be acceptable.

RESPONSE

A natural circulation test program will be performed consisting of several tests to be conducted at low power conditions prior to power escalation and several additional tests performed with the plant at power. Table I.G.1-1 provides a listing of planned tests.

The low power tests will be performed with a critical reactor and will provide operator training in recognizing and evaluating natural circulation conditions and performing certain plant maneuvers while operating under natural circulation conditions. Specific tests will be performed to: 23

- Initiate and verify natural circulation
- Maintain natural circulation with all pressurizer heaters secured
- Demonstrate natural circulation during periods of decaying pressures and
- Demonstrate natural circulation with one steam generator isolated.

These tests will be repeated as necessary to provide training to licensed operators.

Several natural circulation tests will be performed after power operation. In these cases sufficient core decay heat will be available such that a critical reactor will not be required. The tests to be performed during power operation include a loss of offsite power test, a loss of offsite and onsite power test and a test to verify boron mixing and RCS cooldown capability while operating with natural circulation. These tests will provide additional design verification information and supplement operator training.

REFERENCE

None.

TABLE I.G.1-1

NATURAL CIRCULATION TEST PROGRAM

A. LOW POWER TESTING

1. <u>Natural Circulation Verification</u>

This is the first natural circulation test performed. The objective is to provide "baseline" experience to operations personnel.

2. <u>Natural Circulation at Reduced Pressures</u>

Natural circulation is established and maintained with all pressurizer heaters secured. The objective are (1) to provide training on pressure control using charging flow and secondary plant, (2) to demonstrate natural circulation under conditions of decreasing pressure, and (3) to determine the RCS depressurization rate with no pressurizer heaters.

3. <u>Natural Circulation with Reduction of Heat Removal Capacity</u>

This test is designed to show the effect of isolating a steam
23 generator (half of heat removal capacity) and later returning it to service.

B. POWER ESCALATION TESTING

1. <u>Loss of Offsite Power</u>

A reactor trip and natural circulation are initiated by a complete loss of offsite AC power to the plant. This is not a simulation, and emergency diesel power is relied upon. The objective is to verify system behavior under these conditions.

2. <u>Simulated Loss of Offsite and Onsite AC Power</u>

The intent of this test is to demonstrate system operation under conditions of total loss of AC power sources, including the emergency diesel generators. This test will be simulated to a degree in that it is not desirable to secure all plant equipment and risk unnecessary damage.

3. <u>Natural Circulation with Boron Mixing and Cooldown</u>

This test will begin by establishing natural circulation conditions following a trip from power. Following this the RCS will be borated to demonstrate boron mixing capabilities under natural circulation conditions. With the system borated a cooldown will be initiated to bring the plant to a condition where long-term shutdown cooling could be initiated.

II.B.1 - NUREG 0737 REACTOR COOLANT SYSTEM VENTS

REQUIREMENT

Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the events shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information[a] concerning the design and operation of the high point vent system:

1. Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46. 23

2. Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Clarification

A. General

1. The important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensible gas which could interfere with core cooling.

(a) It was the intent of the October 30, 1979 letter to delete the requirement to meet the criteria of 10 CFR 50.44 and SRP 6.2.5 for beyond-design-basis events. The analysis requirements of Position 2 in the September 13, 1979 letter are therefore unnecessary.

2. Procedures addressing the use of the reactor coolant system vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant being able to maintain core cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the normal design basis must not result in a violation of the requirements of 10 CFR 50.44 or 10 CFR 50.46.

3. The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which may be considered, is to specify a volume of noncondensible gas to be vented and in a specific venting time. For containments particularly vulnerable to failure from large hydrogen releases over a short period of time, the necessity and desirability for contained venting outside the containment must be considered (e.g., into a decay gas collection and storage system).

4. Where practical, the reactor coolant system vents should be kept smaller that the size corresponding to the definition of LOCA (10 CFR 50, Appendix A). This will minimize the challenges to the emergency core cooling system (ECCS) since the inadvertent opening of a vent smaller than the
23 LOCA definition would not require ECCS actuation, although it may result in leakage beyond technical specification limits. On PWRs, the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve in series with a vent valve that can be closed from the control room to terminate the LOCA that would result if an open vent valve could not be reclosed.

5. A positive indication of valve position should be provided in the control room.

6. The reactor coolant vent system shall be operable from the control room.

7. Since the reactor coolant system vent will be part of the reactor coolant system pressure boundary, all requirements for the reactor pressure boundary must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures, may be a viable option to meet the single-failure criterion. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10 CFR 50.46.

8. The probability of a vent path failing to close, once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent

isolation of the entire vent system when required. On BWRs, block valves are not required in lines with safety valves that are used for venting.

9. Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

10. The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE 344-1975 as supplemented by Regulatory Guide 1.100, 1.92 and SEP 3.92, 3.43, and 3.10. Environmental qualifications are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21).

11. Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

12. It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

    a. The use of this information by an operator during both normal and abnormal plant conditions,

    b. Integration into emergency procedures, 23

    c. Integration into operator training, and

    d. Other alarms during emergency and need for prioritization of alarms.

B. BWR Design Considerations

1. Since the BWR owners' group has suggested that the present BWR designs have an inherent capability to vent, a question relating to the capability of existing systems arises. The ability of these systems to vent the RCS of noncondensible gas generated during an accident must be demonstrated. Because of differences among the head vent systems for BWRs, each licensee or applicant should address the specific design features of this plant and compare them with the generic venting capability proposed by the BWR owners' group. In addition, the ability of these systems to meet the same requirements as the PWR vent system must be documented.

2. In addition to RCS venting, each BWR licensee should address the ability to vent other systems, such as the isolation condenser which may be required to maintain adequate core cooling. If the production of a

large amount of noncondensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

1. Each PWR licensee should provide the capability to vent the reactor vessel head. The reactor vessel head vent should be capable of venting noncondensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths).

2. Additional venting capability is required for those portions of each hot leg that cannot be vented through the reactor vessel head vent or pressurizer. It is impractical to vent each of the many thousands of tubes in a U-tube steam generator; however, the staff believes that a procedure can be developed that assures sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the RCS. Such operating procedures should incorporate this consideration.

3. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations,
23 especially during natural circulation.

RESPONSE

San Onofre Units 2&3 are in the process of completing the design and installation of a reactor coolant gas vent system (RCGVS) which will be operable prior to fuel load. The system being installed is a vent arrangement from the reactor vessel head and pressurizer remotely operated to either the quench tank or directly to the containment. A description of the design follows:

A. SYSTEM DESIGN BASIS

The reactor coolant gas vent system is designed to remotely vent gases from the reactor vessel head and pressurizer steam space during a post-accident situation where large quantities of noncondensable gases may collect. Although designed for availability during post-accident conditions, the system can also be used to aid in the RCS venting procedures during normal operations.

The design criteria for the RCGVS are as follows:

1. The system permits remote (control room) venting from the reactor vessel head vent or the pressurizer.

2. The vent flow rate capability is based upon the following considerations:

   a. The vent rate is sufficient to vent one-half of the RCS volume in standard cubic feet of H_2 in 1 hour.

   b. Coolant liquid loss through the vent does not exceed makeup capacity.

   c. The vent mass rate does not result in heat loss from the RCS in excess of the normal pressurizer heater capacity. (When only 1E powered heaters are available, the venting will not result in uncontrollable pressurizer pressure or level changes).

3. The system design meets the requirements of 10 CFR 50, Appendix A. Each vent path contains an orifice which restricts mass flow from a break in the piping to less than the definition of a LOCA.

4. Positive indication and controls are provided in the control room for all power operated valves.

5. The vent path is safety grade meeting the same qualifications as the RCS Quality Class I/II and Seismic Category I.

6. Parallel fail closed power operated valves are IE powered from separate emergency buses. Power is removed during normal operation to minimize the possibility of inadvertent operation. Series isolation is provided in the event that a valve does fail to close. 23

7. The system is capable of venting to the quench tank or directly to upper containment where containment fans assure good mixing.

8. The design minimizes modification to currently designed safety class equipment and piping.

9. The system is operable following all design basis events except those requiring evacuation of the control room, and loss of all power (plant blackout).

10. The system is designed to vent superheated steam, steam/water mixtures, water, fission gases, helium, nitrogen, and hydrogen as high as 2500 lb/in^2a and 700F.

11. The system is designed to not interfere with refueling maintenance actions.

12. The system is seismically and environmentally qualified. The solenoid valves are qualified to IEEE-382-1072 for inside containment, IEEE-344-1975 for seismic and IEEE-323-1974 for environmental qualification.

13. Test connections have been included in the design to allow for operability testing of the system.

B. SYSTEM DESCRIPTION

1. System Parameters

The components in the RCGVS consist of piping, valves, and instrumentation to direct vented gas flow from existing components to an existing tank. The main design parameters for the vent line are:

Flow	100 Standard ft^3/min of H_2
Design Temperature	700F
Design Pressure	2500 lb/in^2 a
Line Size	1 inch

2. Descriptive Summary

23 Referring to the schematic flow diagram (figure II.B.1-1), the system permits the control room operator to remotely vent the pressurizer and reactor vessel head. No other high points exist in the reactor coolant system except for the U-tubes in the steam generator which are impractical vent points. The pressurizer vent ties into an existing 3/4-inch pressurizer vapor space sample line. The reactor vessel head vent ties into the existing 3/4-inch reactor vessel vent and will be flanged to permit head removal for refueling. The vent line terminates in the quench tank, entering under the quench volume. The capability to vent directly to containment is also provided when large quantities of gas must be vented. This line terminates in the containment air at a location where good mixing with the containment air is provided and cooling of vented gases is provided. The normal vent paths are from either the pressurizer or reactor vessel head to the quench tank. The vent flow is directed into the quench tank under water to remove energy from the steam which will be vented along with the gas and to cool the gas itself. If large quantities of gas must be vented, the quench tank will pressurize and eventually rupture its rupture disc providing a path to the containment. Venting can be shifted directly to the containment upon actuation of the tank high pressure alarm. Hydrogen concentration will be controlled by the containment hydrogen recombiners. See FSAR Subsection 6.2.5 for a discussion of combustible gas control in containment. The vent paths are seismically designed and are of the appropriate quality class. All valves are powered from emergency power sources. Parallel valves provide redundancy where necessary to provide assurance that the system will operate under accident conditions. For cases when the vented gas is not highly radioactive and containment isolation has not occurred, the gas delivered to the quench tank is

routed through the gaseous waste management system (GWMS) containment vent header to the gas surge tank. This path will allow processing of gas removed by the vent system if the system is used to aid in RCS venting either prior to or following refueling. Although designed for accident conditions, the system may be used to aid in pre- or post-refueling venting of the RCS. Pressure instrumentation is included in the design to monitor system performance.

3. Operation

Operator action to correct plant conditions causing noncondensable gas generation depends upon the severity of the event and the auxiliary services available for response.

The operator can vent noncondensable gases from the reactor coolant via the following flow paths:

- Reactor vessel head or pressurizer vent to the quench tank
- Reactor vessel or pressurizer vent to containment directly
- Reactor vessel head or pressurizer vent to the gaseous waste management system

Operating considerations for each flow path are given below: 23

a. Reactor Vessel Head or Pressurizer Vent to the Quench Tank

Venting to the quench tank is accomplished by opening either of the parallel reactor vessel head vent isolation valves or pressurizer vent isolation valves and the quench tank isolation valves. Vent line pressure instrumentation provides indication of system performance. The vent flow is dependent upon RCS pressure and a timed vent will be based on estimated gas volume. Integrity of the quench tank is confirmed by tank pressure, temperature, and level.

b. Reactor Vessel Head or Pressurizer Vent to Containment Directly

Extended venting from either source to the quench tank may eventually rupture the quench tank rupture disc. While this may be of no consequence during a major accident, there may be times when the operator would desire to maintain the rupture disc intact. In this event, the venting may be directed to the containment directly by opening the containment vent isolation valve.

c. Reactor Vessel Head Pressurizer Vent to the Gaseous Waste Management System (GWMS)

This path may be used if the gases are known to be low in activity, (i.e., will not exceed GWMS Tech. Spec. values), normal power is available, low removal rates are acceptable, the containment is not isolated, and adequate storage space is available in the GWMS. Alignment of this path is accomplished by initiating vent flow from the pressurizer or reactor vessel head to the quench tank. Gases directed to the quench tank are then removed from the tank via the containment vent header of the GWMS and are collected in the gas surge tank for eventual storage in the gas decay tanks.

Consistent with the NRC requirements, procedures for system use are being provided. The vent system should not be used unless there is a bubble to be removed. The system will utilize on-off control to achieve the overall vent rate desired. Estimated vent flow rate as a function of RCS pressure is being provided to the operator, as will estimated venting times based upon gas volume. Since venting can be terminated at any time by the operator, if the requirements of 10 CFR 50.44 or 10 CFR 50.46 are approached, venting can be stopped.

C. SYSTEM CONTROLS

23 The system is designed to be controlled remotely from the main control room. All valves and instrumentation are powered from emergency power sources and alternate sources are used as necessary to meet single failure criteria. Position indication (open/shut) is provided for all remotely operated valves and is displayed in the control room. Pressure instrumentation is provided to monitor system performance and is displayed in the control room.

D. LOCA ANALYSIS

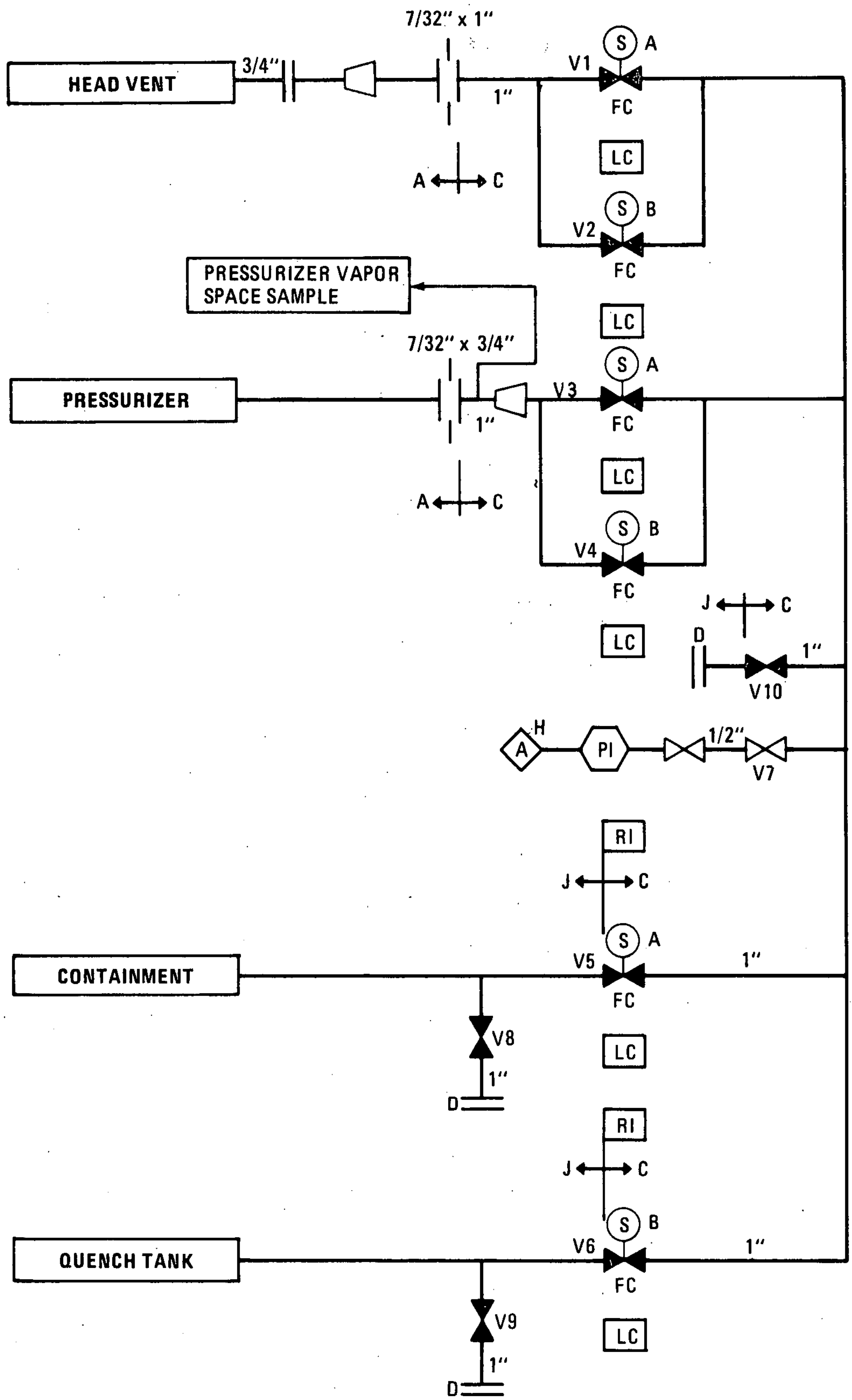
Consistent with NRC requirements, the system is designed to limit mass loss to less than a LOCA as defined in 10 CFR 50, Appendix A, and thus a separate analysis of inadvertent system operation or pipe breakage is not required. A failure mode and effects analysis is provided in FSAR table 9.3-18.

E. DOCUMENTATION

Section 9.3.7 has been added to the FSAR to describe the Reactor Coolant Gas Vent System. Included in this section are discussions of design bases, design criteria, controls, piping and arrangements, system description, safety evaluation, inspection testing requirements, and instrumentation requirements. Electrical schematics are provided as noted in FSAR table 1.7-1. There are no associated control logic diagrams.

REFERENCE

FSAR subsections 6.2.5 and 9.3.7. 23



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

REACTOR COOLANT GAS VENT SYSTEM SKETCH

Figure II.B.1-1

II.B.2 - NUREG 0737 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST-ACCIDENT OPERATIONS

REQUIREMENT

Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas of protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall 23
determine which types of corrective actions are needed for vital areas throughout the facility.

Clarification

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short-term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. The actions should be taken pending conclusions resulting in the long-term degraded core rulemaking, which may result in a need to consider additional sources.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

The control room, technical support center (TSC), sampling station and sample analysis area must be included among those areas where access is considered vital after an accident. (Refer to section III.A.1.2 for discussion of the TSC and emergency operations facility.) The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be for these areas if they are determined not to be vital.

As a minimum, necessary modifications must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

In order to assure that personnel can perform necessary post-accident operations in the vital areas, the following guidance is to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. <u>Source Term</u>

   The minimum radioactive source term should be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, 1.7 and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., assuming the radioactive decay that occurs before fission products can be transported to various systems).

   a. <u>Liquid-Containing Systems</u>: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% of all others are assumed to be mixed in the reactor coolant and liquids recirculated by residual heat removal (RHR), high-
23 pressure coolant injection (HPCI), and low-pressure coolant injection (LPCI), or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, assuming that the water contains no noble gases.

   b. <u>Gas-Containing Systems</u>: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor-containing lines connected to the primary system (e.g., BWR steam lines), the concentration of radioactivity shall be determined assuming the activity is contained in the vapor space in the primary coolant system.

2. <u>Systems Containing the Source</u>

   Systems assumed in your analysis to contain high levels of radioactivity in a post-accident situation should include, but not be limited to, containment, residual heat removal system, safety injection systems, chemical and volume control system (CVCS), containment spray recirculation system, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, you should explain why such systems were excluded. Radiation from leakage of systems located outside of containment need not be considered for this analysis. Leakage measurement and reduction is treated under section III.D.1.1, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and BWRs." Liquid waste systems need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of section III.D.1.4, "Radwaste System Design Features To Aid in Accident Recovery and Decontamination."

3. Dose Rate Criteria

The design dose rate for personnel in a vital area should be such that the guidelines of GDC 19 will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case bases. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and are not to be used to limit access in the event of an accident. 23

a. Areas Requiring Continuous Occupancy: <15 mrem/hr (averaged over 30 days). These areas will require full-time occupancy during the course of the accident. The control room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in Standard Review Plan 6.4.

b. Areas Requiring Infrequent Access: GDC 19. These areas may require access on an irregular basis, not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant radiochemical/chemical analysis laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples of sites where occupancy may be needed often, but not continuously.

4. Radiation Qualification of Safety-Related Equipment

The review of safety-related equipment which may be unduly degraded by radiation during post-accident operation of this equipment relates to equipment inside and outside of the primary containment. Radiation source terms calculated to determine environmental qualification of safety-related equipment consider the following:

a. LOCA events which completely depressurize the primary system should consider releases of the source term (100% noble gases, 50% iodines, and 1% particulates) to the containment atmosphere.

b. LOCA events in which the primary system may not depressurize should consider the source term (100% noble gases, 50% iodines, and 1% particulates) to remain in the primary coolant. This method is used to determine the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment. Non-LOCA events both inside and outside of containment should use 10% noble gases, 10% iodines, and 0% particulate as a source term. The following table summarizes these considerations:

23

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High-Energy Line Break Source Term (Noble Gas/Iodine/Particulate)
Outside	% (100/50/1) in RCS	% (10/10/0) in RCS
Inside	Larger of (100/50/1) in containment or (100/50/1) in RCS	(10/10/0) in RCS

RESPONSE

A design review of plant radiation and shielding for post-accident operations has been performed. The review considered the potential radiation exposure to operators and equipment in vital areas. The following sections describe the assumptions and methodology employed in the review.

1.0 SOURCE TERMS

The source terms used in the evaluation of the shielding are based on the postulated post-accident release of radioactivity equivalent to that described in Regulatory Guide 1.4 and are further detailed in the four sections below.

1.1 CORE INVENTORY SOURCES

Table II.B.2-1 presents the core inventory of radioisotopes. These sources are based on 105% of full power core conditions (3560 Mwt).

1.2 REACTOR COOLANT

The following fractions of core inventory are diluted in the reactor coolant volume of 82,000 gallons.

Table II.B.2-1

CORE INVENTORY (CURIES)

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Br-84	2.85(7)	Mo-99	1.89(8)	I-135	1.89(8)
Br-85	3.99(7)	Tc-99m	2.27(7)	Xe-135m	5.51(7)
Kr-85m	3.98(7)	Ru-103	9.18(7)	Xe-135	4.62(7)
Kr-85	8.73E5	Ru-106	8.23(6)	Cs-135	1.15(1)
Kr-87	7.44(7)	Te-129m	1.07(7)	Cs-136	1.78(5)
Kr-88	1.09(8)	Te-129	3.29(7)	Xe-137	1.80(8)
Rb-88	1.10(8)	I-129	2.08(0)	Cs-137	4.15(6)
Kr-89	1.41(8)	I-131	8.96(7)	Xe-138	1.79(8)
Rb-89	1.46(8)	Xe-131m	6.16(5)	Cs-138	2.04(8)
Sr-89	1.45(8)	Te-132	1.33(8)	Cs-140	1.81(8)
Sr-90	9.27(6)	I-132	1.33(8)	La-140	1.95(8)
Y-90	9.22(6)	Te-133m	1.07(8)	Ba-143	1.60(8)
Sr-91	1.78(8)	Te-133	1.13(8)	La-143	1.81(8)
Y-91m	1.05(8)	I-133	2.06(8)	Ce-143	1.81(8)
Y-91	1.79(8)	Xe-133	1.97(8)	Pr-143	1.81(8)
Y-95	1.87(8)	Cs-134	1.03(6)	Ce-144	1.26(8)
Zr-95	1.87(8)	Te-134	2.12(8)	Pr-144	1.26(8)
Nb-95	1.90(8)	I-134	2.39(8)		

- 100% Noble Gas
- 50% Halogens
- 1% Solids

Table II.B.2-2 presents reactor coolant concentrations at start of postulated accident.

Table II.B.2-2

REACTOR COOLANT CONCENTRATIONS (CURIES/CC)
(AT START OF ACCIDENT)

Nuclide		Nuclide		Nuclide	
BR-84	4.59E-02	RU-106	2.64E-04	XE-135	1.48E-01
BR-85	6.43E-02	TE-129M	3.44E-04	CS-135	3.69E-10
KR-85M	1.27E-01	TE-129	1.05E-03	CS-136	5.72E-06
KR-85	2.80E-03	TC-99M	7.29E-04	XE-137	5.78E-01
KR-87	2.39E-01	I-129	3.34E-09	CS-137	1.33E-04
KR-88	3.50E-01	I-131	1.44E-01	XE-138	5.75E-01
RB-88	3.53E-03	XE-131M	1.98E-03	CS-138	6.55E-03
Kr-89	4.54E-01	TE-132	4.27E-03	CS-140	5.81E-03
RB-89	4.69E-03	I-132	2.14E-01	LA-140	6.26E-03
SR89	4.66E03	TE-133M	3.44E-03	BA-143	5.14E-03
SR-90	2.98E-04	TE-133	3.63E-03	LA-143	5.81E-03
Y-90	2.96E-04	I-133	3.31E-01	CE-143	5.81E-03
SR-91	5.72E-03	XE-133	6.33E-01	PR-143	5.81E-03
Y-91M	3.37E-03	CS-134	3.31E-05	CE-144	4.05E-03
Y-91	5.75E-03	TE-134	6.81E-03	PR-144	4.05E-03
NB-95	6.10E-03	I-134	3.85E-01	ZR-95	6.01E-03
MO-99	6.07E-03	I-135	3.03E-01		
RU103	2.95E03	XE-135M	1.77E-01		

1.3 CONTAINMENT ATMOSPHERE

The following fractions of core inventory are diluted in the containment volume of 2.3 x 10^6 ft^3.

- 100% Noble Gas
- 25% Halogens
- 1% Solids

Table II.B.2-3 presents containment airborne concentrations at start of postulated accident.

Table II.B.2-3

CONTAINMENT AIRBORNE CONCENTRATIONS (CURIES/CC)
(AT START OF ACCIDENT)

Nuclide		Nuclide		Nuclide		
BR-84	1.06E-04	RU-106	1.23E-06	XE-135	6.93E-04	
BR-85	1.49E-04	TE-129M	1.60E-06	CS-135	1.72E-12	
KR-85M	5.97E-04	TE-129	4.93E-06	CS-136	2.67E-08	
KR-85	1.30E-05	I-129	7.80E-12	XE-137	2.70E-03	
KR-87	1.11E-03	I-131	3.36E-04	CS-137	6.22E-07	
KR-88	1.63E-03	XE-131M	9.24E-06	XE-138	2.68E-03	
KR-89	2.12E-03	TE-132	1.99E-05	CS-138	3.06E-05	
RB-88	1.65E-05	I-132	4.99E-04	CS-140	2.71E-05	21
RB-89	2.19E-05	TE-133M	1.60E-05	LA-140	2.92E-05	
SR-89	2.17E-05	TE-133	1.69E-05	BA-143	2.40E-05	
SR-90	1.39E-06	I-133	7.72E-04	LA-143	2.71E-05	
Y-90	1.38E-06	XE-133	2.95E-03	CE-143	2.71E-05	
SR-91	2.67E-05	CS-134	1.54E-07	PR-143	2.71E-05	
Y-91M	1.57E-05	TE-134	3.18E-05	CE-144	1.89E-05	
Y-91	2.68E-05	I-134	8.97E-04	PR-144	1.89E-05	
NB-95	2.85E-05	I-135	7.09E-04	ZR-95	2.80E-05	
MO-99	2.83E-05	XE-135M	8.26E-04	TC-99M	3.40E-06	
RU-103	1.37E-05					

1.4 CONTAINMENT SUMP

The following fractions of core inventory are diluted in the sump volume of 400,000 gallons.

- 100% Noble Gas
- 50% Halogens
- 1% Solids

Table II.B.2-4 presents sump concentrations at start of recirculation mode of operation (30 minutes after start of accident).

Table II.B.2-4

CONTAINMENT SUMP CONCENTRATIONS (CURIES/CC)
(30 MINUTES AFTER START OF ACCIDENT)

Nuclide		Nuclide		Nuclide	
BR-84	9.53E-06	RU-106	5.48E-08	XE-135	3.08E-05
BR-85	1.33E-05	TE-129M	7.13E-08	CS-135	7.67E-14
KR-85M	2.65E-05	TE-129	2.19E-07	CS-136	1.18E-09
KR-85	5.82E-07	I-129	6.93E-13	XE-137	1.20E-04
KR-87	4.96E-05	I-131	2.98E-05	CS-137	2.76E-08
KR-88	7.27E-05	XE-131M	4.10E-07	XE-138	1.19E-04
RB-88	7.33E-07	TE-132	8.87E-07	CS-138	1.36E-06
RB-89	1.91E-06	I-132	4.44E-05	CS-140	1.20E-06
SR-89	9.67E-07	TE-133M	7.13E-07	LA-140	1.30E-06
SR-90	6.18E-08	TE-133	7.53E-07	BA-143	1.06E-06
Y-90	6.14E-08	I-133	6.87E-05	LA-143	1.20E-06
SR-91	1.18E-06	XE-133	1.31E-04	CE-143	1.20E-06
Y-91M	7.00E-07	CS-134	6.87E-09	PR-143	1.20E-06
Y-91	1.19E-06	TE-134	1.41E-06	CE-144	8.40E-07
NB-95	1.26E-06	I-134	8.00E-05	PR-144	8.40E-07
MO-99	1.26E-06	I-135	6.30E-05	ZR-95	1.24E-06
RU-103	6.12E-07	XE-135M	3.67E-05	TC-99M	1.51E-07

2.0 SYSTEM REVIEW

After establishing the above source terms, the systems outside containment which may contain highly radioactive materials have been evaluated. The specific systems reviewed are described in the following six sections. As a result of this review, the addition of one steel shielding door was required. The detailed results of the system review described in this section are provided in section 4.0.

2.1 SAFETY INJECTION SYSTEM

Plant shielding has been evaluated considering the low-pressure safety injection pumps to be operating in the shutdown cooling mode. The source terms for this operation are discussed in section 1.2.

For purposes of plant shielding review, the high-pressure safety injection pumps are assumed to be operating in the recirculation mode. Prior to start of recirculation, the high-pressure safety injection system will contain non-radioactive water from the refueling water storage tank. At the start of recirculation (30 minutes after start of accident), the source terms of section 1.4 are used.

2.2 CONTAINMENT SPRAY SYSTEM

Plant shielding has been evaluated considering the containment spray system to be operating in the recirculation mode. Prior to the start of recirculation, the containment spray system will contain non-radioactive water from the refueling water storage tank. At the start of recirculation (30 minutes after start of accident), the source terms of section 1.4 are used.

2.3 REACTOR COOLANT CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS)

The CVCS is not considered to contain highly radioactive fluid following a postulated accident, except for piping used in the post-accident sampling system return line. This system is not assumed to become highly radioactive because:

A. The system is automatically isolated.

B. The letdown system is not required for accident mitigation.

C. Post-accident degassing capability of primary system is provided by remotely operated reactor coolant high point vent system (refer to section II.B.1).

2.4 COOLANT RADWASTE SYSTEM

The coolant radwaste system is not assumed to contain highly radioactive fluid following a postulated accident, except for the piping used in the sample return line. This assumption is based on:

A. The system is automatically isolated.

B. The system is not required for accident mitigation.

The piping used in the sample liquid return line (refer to section II.B.3) is considered to contain source terms of section 1.2.

2.5 WASTE GAS SYSTEM

The waste gas surge and decay tanks are not considered to contain highly radioactive material generated following an accident, since:

A. The waste gas system is automatically isolated.

B. The waste gas system is not required for accident mitigation.

C. Post-accident degassing capability of primary system is provided by remotely operated reactor coolant high point vent system (refer to section II.B.1).

D. Waste gas generated in the sample lab is returned to the containment (refer to section II.B.3).

The piping used in the sample gas return line (refer to section II.B.3) is considered to contain highly radioactive material.

2.6 NUCLEAR PLANT SAMPLING SYSTEM

Section II.B.3 discusses the design features of the sample lab used for highly radioactive fluid.

The lines to the post-accident sample lab are considered to contain sources discussed in section 1.0 consistent with the intended service. The portions of the sample lines between the post-accident and normal operation sample labs are not considered to contain highly radioactive material since automatic isolation prevents this condition (refer to section II.B.3).

3.0 SHIELDING METHODS

Evaluation of plant shielding following an accident includes direct radiation from containment along with radiation from piping and components of systems discussed in section 2.0. Direct, scatter, and radiation streaming through penetrations are considered. For example, in evaluating the doses at the wide range effluent monitor location, contributions from streaming through the containment personnel lock and the purge penetration along with subsequent scatters were calculated.

FSAR paragraph 12.3.2.3 discusses the analytical methods used in the shielding analysis.

4.0 DESIGN REVIEW

The plant shielding design review identifies the radiation exposure to vital areas requiring personnel access following an accident. The plant modifications, if any, associated with the system review described in section 2 are described in the following paragraphs:

23 The integrated dose to the control room, technical support center, and security central alarm station is less than 5 Rem wholebody for the duration of the accident in accordance with GDC 19. As part of the shielding review, an 8.75-inch steel shield door was added, as shown in figures II.B.2-1 and II.B.2-2. This steel door reduces streaming through the passageway from piping in the penetration area.

The present normal operation sample lab is not shielded to accommodate highly radioactive fluid. To meet the requirements for post-accident sampling, a new system, described in section II.B.3, is being added. The layout of the sample lab is still being developed. Figure II.B.2-2 shows the room being used for the post-accident sample lab. The criteria established in section II.B.3 will be used to determine the specific shielding arrangement. In addition, additional shielding has been added in a corridor to reduce the access dose rates from the sample line pipechase.

Operations required to allow post-accident operation of the shutdown cooling system include manual operation of certain valves. These operations must be completed prior to initiation of recirculation because of the high radiation levels in the ECCS pump rooms following recirculation. The SONGS 2 and 3 project is reviewing procedures and methods to minimize the potential for operator exposure during alignment of this system for shutdown cooling. The results of this review will be transmitted to the NRC in an FSAR amendment by approximately February 1981.

The wide-range effluent monitors (refer to section II.F.1) have been designed to minimize operator exposures. Controls for filter, grab sample, and flow switching controls are located in the control room. The monitors are located to facilitate access for removal of sample cartridges. The sample tubing is routed to minimize crud traps.

Post-accident radiation zone drawings have been developed to aid in evaluation of operator exposures and in the writing of emergency procedures. These drawings are FSAR figures 12.3-35, through 12.3-61. Based on these drawings, it can be identified that the motor control centers, instrument panels, emergency power supplies, batteries, and ESF switchgear located in the control building will be accessible following an accident. As described in sections 2.3, 2.4, 2.5; the letdown system and coolant and gaseous radwaste systems are not required for accident mitigation. For this reason, access to the radwaste control panels is not considered to be vital following an accident. 23

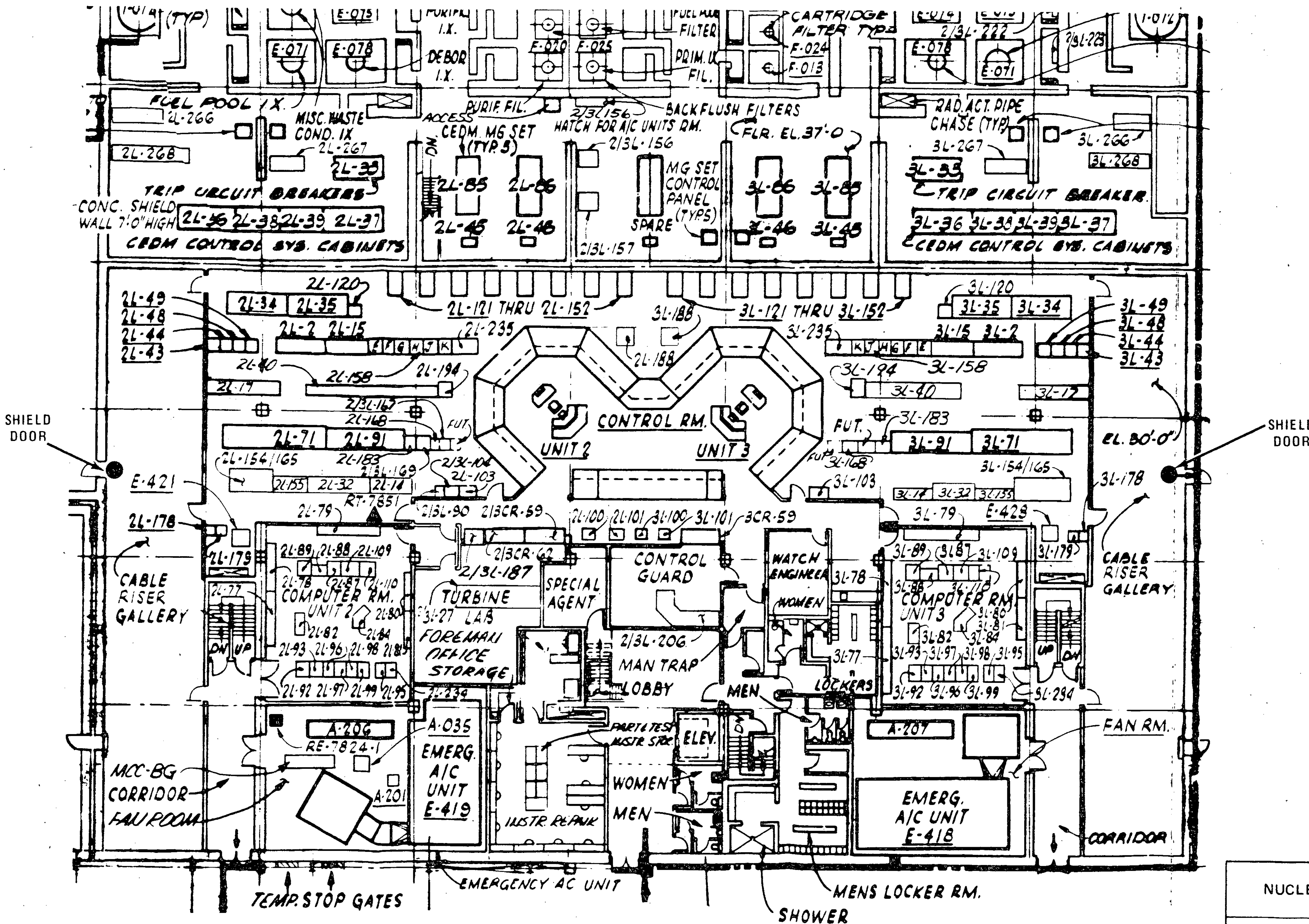
The environmental qualification of equipment at SONGS 2/3 is being done in accordance with the guidelines of NUREG-0588.

5.0 MEETING NRC REQUIREMENTS

The design review of plant shielding discussed herein fulfills the NRC requirements as outlined in NUREG-0578, Clarification to NUREG-0578 (NRC Letter, Nov. 9, 1979), NUREG-0660, NUREG-0694 and NUREG-0737. 23

6.0 REFERENCES

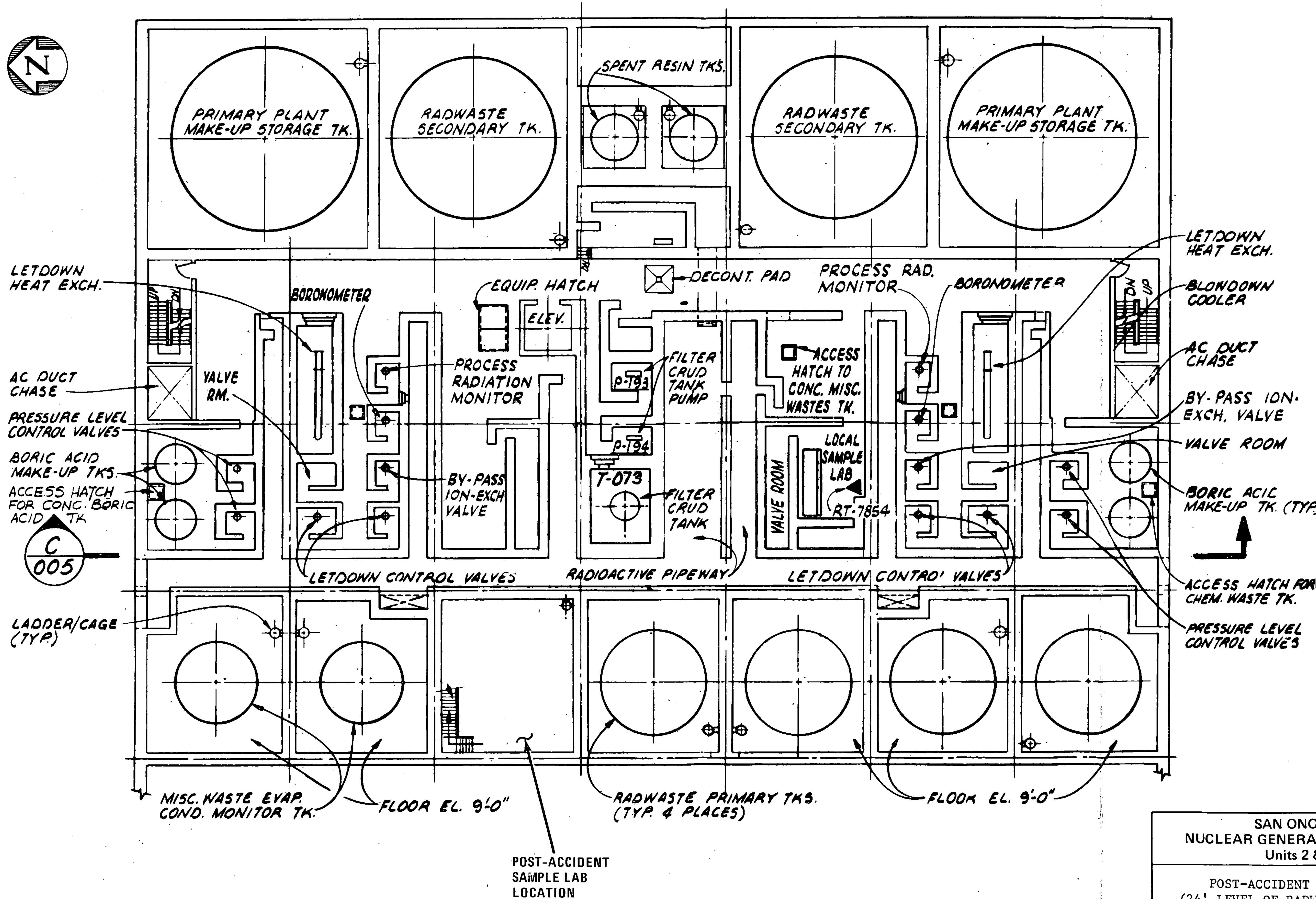
Refer to revised FSAR chapter 12, Radiation Protection, and chapter 15, Transient Analysis.



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CONTROL ROOM
STEEL SHIELDING DOOR

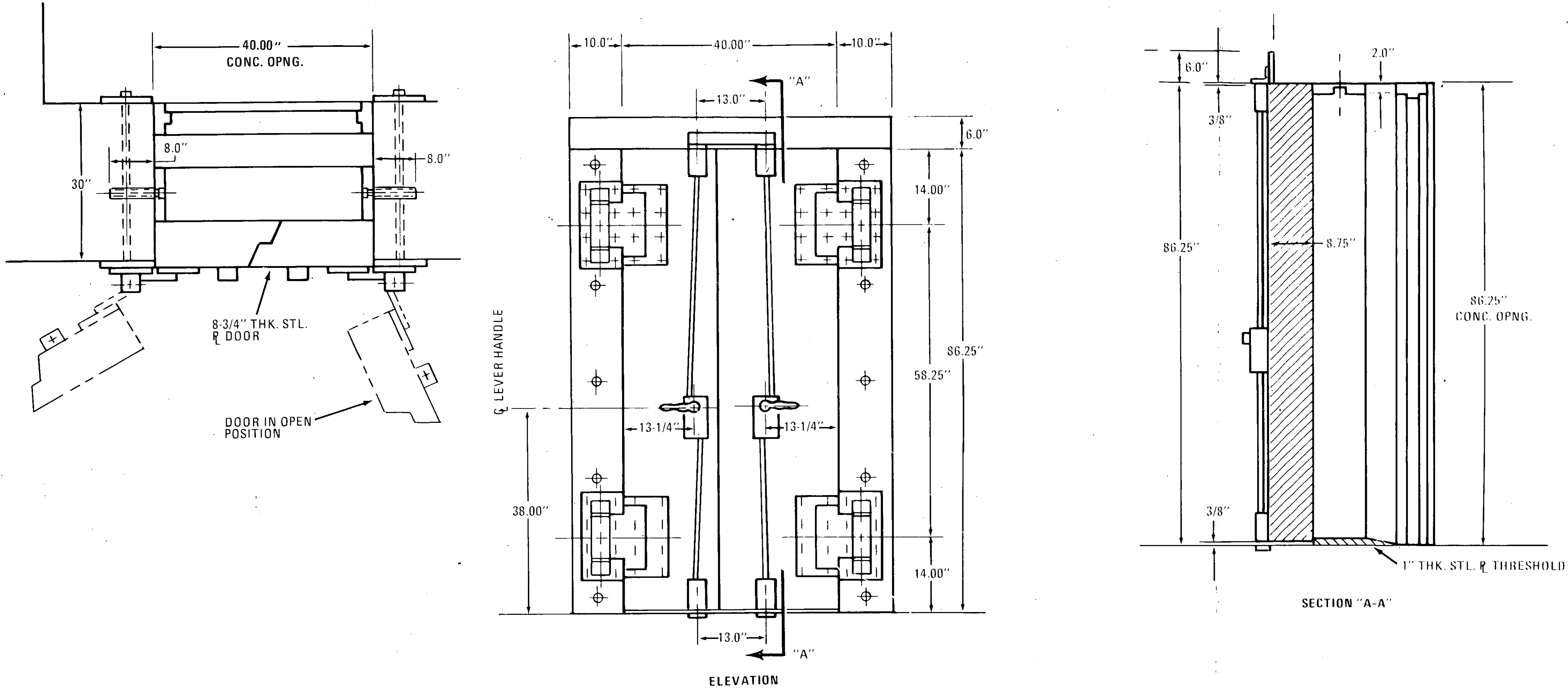
Figure II.B.2-1



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

POST-ACCIDENT SAMPLE LAB
(24' LEVEL OF RADWASTE BUILDING)

Figure II.B.2-2



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CONTROL ROOM
STEEL SHIELDING DOOR
DETAILS

Figure II.B.2-3

II.B.3 - NUREG 0737 POST-ACCIDENT SAMPLING CAPABILITY

REQUIREMENT

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible 23
contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Clarification

The following items are clarifications of requirements identified in NUREG-0578, NUREG-0660, or the September 13 and October 30, 1979 clarification letters.

1. The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

2. The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

   a. Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and nonvolatile isotopes);

   b. Hydrogen levels in the containment atmosphere;

   c. Dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

   d. Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

3. Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

23

4. Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

5. The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish waster and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within four days. The chloride analysis does not have to be done onsite.

6. The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). [Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).]

7. The analysis of primary coolant samples for boron is required for PWRs. (Note that Revision 2 of Regulatory Guide 1.97, when issued, will likely specify the need for primary coolant boron analysis capability at BWR plants.)

8. If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for seven days following onset of the accident and at least one sample per week until the accident condition no longer exists.

9. The licensee's radiological and chemical sample analysis capability shall include provisions to:

   a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g. 23

   b. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

10. Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

11. In the design of the post-accident sampling and analysis capability, consideration should be given to the following items:

   a. Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and

containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be
23 taken from containment. The residues of sample collection should be returned to containment or to a closed system.

b. The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

RESPONSE

1.1 GENERAL DESCRIPTION

FSAR subsection 9.3.2 has a detailed description of the existing process sampling system presently installed in San Onofre Units 2 & 3. The following additional sampling system is being added to specifically address the NRC requirements for post-accident sampling (NUREG-0578, Clarification to NUREG-0578, NUREG-0660, NUREG-0694, NUREG 0737). This new emergency sampling system will be installed prior to operation above 5% power.

A review of the reactor coolant and containment atmosphere sampling systems and the radiological spectrum and chemical analysis facilities has been conducted. Plant modifications are being implemented to permit personnel to obtain and analyze samples within 3 hours after a decision is made to take a sample (without incurring an exposure to an individual in excess of five rem whole-body or 75 rem to the extremities). Provisions will be made to allow a chloride analysis within four days. These plant modifications, collectively referred to as the post-accident sampling system (PASS), are illustrated schematically by figures II.B.3-1,
23 II.B.3-2, II B.3-3 and II.B.3-4.

Procedures will be developed for obtaining and analyzing these samples.

Operation of the PASS does not require isolated auxiliary systems to be placed in operation. However, portions of piping in the isolated containment waste gas header and the reactor coolant drain tank header are utilized to return sample effluent fluids and gasses to the containment. These sections of piping are included in the leak-test program described in item III.D.1.1.

1.2 DESIGN CRITERIA

Modification of currently existing plant facilities allows plant personnel to obtain and analyze pressurized and unpressurized reactor coolant samples and a containment air sample within 3 hours.

Onsite facilities and procedures are being developed which provide the
23 capability to quantify the following:

A. Certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines, cesiums, and non-volatile isotopes),

B. Hydrogen levels in the containment atmosphere in the range of 0 to 10 volume %.

C. Dissolved gases (i.e., H_2, O_2) and boron concentration of liquids.

The onsite facility design and procedures satisfy the following criteria:

A. Provisions are included to permit containment atmosphere sampling under both positive and negative containment pressure.

B. Provisions are included for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for passive flow restrictions to limit reactor coolant loss or containment atmosphere leakage from a rupture of the sample line.

C. The PASS sampling lines and components conform to Quality Group D and Seismic Category II requirements. If the classification of the system to which a PASS sample line is attached is higher than the classification of the PASS, then the PASS piping up to the first isolation valve meets the higher classification as shown in figure II.B.3-1.

D. Provisions are included to identify and quantify the isotopes of the nuclide categories discussed above in the range from approximately 1 mCi/g to levels corresponding to an initial reactor coolant radiochemical spectrum resulting from a Regulatory Guide 1.4 (Revision 2) release. Where necessary, sample dilution is used to provide the capability for measurement using onsite installed multi-channel analysis equipment and for reduction of personnel exposure.

E. Provisions are included to measure total dissolved gas concentrations up to approximately 2000 cc/kg.

F. Provisions are included to restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis provides results with an acceptably small error (approximately a factor of 2). This is accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design, including a charcoal filter and HEPA filters which will control the presence of airborne radioactivity. 23

G. Provisions are included to maintain plant procedures which identify the analyses required, measurement techniques, and background level reduction methods.

H. Chemical analysis capability is provided which can obtain the results discussed above in the presence of the radiological source term indicated for the radiological analysis.

I. Provisions are included to make it possible to obtain and analyze a sample while incurring a radiation dose to any individual that is as low as is reasonably achievable and not in excess of General Design Criterion 19. On assuring that these limits are met, facility design and procedures meet the following criteria:

1. For shielding calculations, source terms are the maximum of those given in criterion D above.

2. Access to the sample station and the radiological and chemical analysis facilities is through areas which are accessible in post-accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

3. Operations in the sample station, handling of highly radioactive samples from the sample station to the analysis facilities, and handling while working with the samples in the analysis facilities, are such that radiation dose criteria are met. To this end sufficient shielding of personnel from 1) radioactive sample lines in the sample station, 2) the samples themselves in analysis facilities, and 3) other radioactive lines and tanks in the vicinity of the sampling station and analysis facilities is provided. Capabilities for remote handling and dilution of samples for analysis also are provided.

4. High range portable survey instruments and personnel dosimeters are provided to permit rapid assessment of high exposure rates and accumulated personnel exposure.

J. Design parameters for the system are based on reactor coolant design pressure and temperature of 2485 lb/in^2g and 650F, respectively, and 1 gal/min reactor coolant flow for liquid sampling, 100 lb/in^2g and 300F for the containment atmospheric sampling portion.

K. When plant modifications are completed and procedures implemented, testing will be conducted to demonstrate the capability to obtain and analyze a sample containing the isotopes discussed above according to the criteria listed in this section.

1.3 DETAILED DESCRIPTION

1.3.1 SAMPLE INLET PIPING

All PASS samples are drawn from existing sample lines outside containment. Samples are drawn from the following four points in each unit:

A. RCS hot leg No. 1/hot leg No. 2 normal sample line 23

B. RCS pressurizer surge line normal sample line

Both of these samples are normally cooled just downstream of the containment penetrations. In order to ensure, under post-accident conditions, that representative samples reach the PASS sample station, valves are added which allow these coolers to be bypassed post-accident. Between these coolers and the normal process sampling station, these normal sample lines are routed through a pipe chase which is directly adjacent to an available room in the 24-foot level of the radwaste building selected to house most of the equipment comprising the PASS. Each of these two normal sample lines is routed into the PASS room and branched through two stop valves, one to the PASS sample station and the other back out of the room to the normal process sampling station. In order to protect the normal sampling station from the high level source terms present immediately following an accident (which may occur while normal sampling is in progress), each stop valve on the branch going to the normal sampling station is shut automatically by dual range area radiation monitors viewing these lines where they are routed through the PASS room. 23

C. Containment sump via the ESF pump miniflow recirculation line sample line

Contamination of the normal sample station due to contamination in this line will also be prevented by the area radiation monitors in the same manner as sample lines A and B above.

D. Containment Atmosphere Via the Airborne Radiation Monitor Suction Line

The PASS containment atmosphere sample line is connected to the containment airborne radiation monitor suction line between the containment penetration and the outer containment isolation valve. An additional containment isolation valve is provided for the PASS containment atmosphere sample line. In order to assure that a representative sample reaches the PASS sample station, this line is insulated and heat traced so that containment temperature is maintained in the sample until it reaches the PASS sample station. 23

1.3.2 SAMPLE RETURN PIPING

In order to minimize the amount of radioactivity outside containment post-accident, all purged fluids and all liquid and gaseous sample waste is returned from the PASS sample station to the containment through existing containment penetrations. Liquids are pumped back to either the reactor coolant drain tank or the containment normal sump, at the option of sampling personnel, through the existing reactor coolant drain tank discharge header. A stop valve has been added in the reactor coolant drain tank discharge header to allow isolation of the normal coolant
23 radwaste system inlet from the PASS. Gases are returned to the containment atmosphere through the existing waste gas header. A stop valve has been added to the waste gas header to allow isolation of the normal waste gas surge tank inlet from the PASS.

1.3.3 SAMPLE STATION

The sample station is a free-standing cabinet, floor-mounted in a shielded enclosure, containing the following sample collection and analyses equipment as illustrated schematically in figures II.B.3-2 and II.B.3-3:

A. Reactor Coolant Sampling Equipment

1. Piping and valves for drawing, purging, cooling, degassing, circulating, and diluting reactor coolant samples and for routing liquid and gaseous sample waste back to containment

2. Online instruments for the measurement of boron concentration and pH

3. An online germanium detector and remotely located multichannel analyzer for isotopic analysis

4. A sample vessel/heat exchanger, stainless steel burette, and positive displacement pump for cooling, depressurization, degassing, and measurement of sample total gas concentration

5. Online instruments for measurement of hydrogen and oxygen concentrations in the evolved gases

6. A gas sample vial for nitrogen dilution and subsequent grab sampling of the diluted evolved gases

7. A depressurized liquid sample vial for dilution and grab sampling liquid samples, utilizing a long needle syringe

8. A surge vessel and positive displacement surge vessel pump to allow for holdup and return of sample waste to the containment

9. Online flow, pressure, and temperature instrumentation.

10. Provision for obtaining undiluted liquid grab samples in a shielded sample container (figures II.B.3-1 and -4). 23

B. Containment Atmosphere Sampling Equipment

1. Piping and valves for drawing, purging, and diluting containment atmosphere samples and for routing sample waste back to containment

2. An air pump for drawing and returning the sample

3. Online instrumentation for measurement of hydrogen gas concentration in the sample

4. An online germanium detector and remotely located multichannel analyzer for isotopic analysis

5. A vessel for nitrogen dilution and subsequent grab sampling of the diluted containment atmosphere utilizing a long needle syring

6. Online flow instrumentation.

7. Provision for obtaining undiluted gas grab samples in a shielded sample container (figures II.B.3-1 and -4). 23

1.3.4 CONTROL PANEL

The PASS control panel is a free-standing cabinet in the PASS room, shielded from the sample station, containing the following equipment to provide complete control of the PASS system to the operator:

A. Instrumentation readouts with recorders, and valve position indication.

B. Valves, handswitches and pump control.

C. Germanium detector controls.

In addition, a PASS mimic control panel is provided. This is a graphic panel mapping the lines between the PASS and Units 2 and 3 containments and liquid and gaseous radwaste systems. Hand switches and valve position indicators are provided for the PASS valves not included in the basic sample station and shown on figure II.B.3-1. Position indication only is provided on the mimic panel for the PASS-related containment isolation valves and PASS valves inside containment which are operated from the main control room. A separate small panel is located near the PASS control and mimic control panels to provide heat tracing controls for the containment air sample lines. 23

1.3.5 PASS SAMPLE LAB FACILITY

Most of the PASS sample piping and valves, the sample station, control
panel, and other auxiliary equipment are housed in the PASS sample room
23 located on the 24-foot level of the radwaste building. The equipment
arrangement in the PASS room is shown in figure II.B.3-4. Refer to
section II.B.2 for a description of the radiation shielding which is pro-
vided for this room. The following services are provided to this lab
facility:

A. To The Sample Station

23 1. Component cooling water for sample cooling

2. Demineralized water for sample dilution and flushing

3. Pressurized nitrogen gas for gas dilution and purging

4. Instrument air for pneumatically controlled valves

B. An emergency decontamination shower and eyewash for the safety of sampling personnel

C. A sample station vent with a charcoal filter and a room exhaust
23 HEPA filter.

D. Room air conditioning.

1.3.6 CIAS OVERRIDE CAPABILITY

In order to permit post-accident sampling, the capability is provided at a
control room handswitch panel to override the CIAS signal to 12 valves
23 in each reactor unit: HV-7800, HV-7801, HV-7816, HV-0508, HV-0509, HV-0512,
HV-0513, HV-0517, HV-7512, HV-7513, HV-7258, and HV-7259, as shown in
figure II.B.3-1. A description of the CIAS override capability is given in
section II.E.4.2.

1.3.7 PASS SYSTEM OPERATION

Operator action to collect and analyze reactor coolant and containment atmosphere samples during post-accident conditions consists of (1) remote operation of valves and components and (2) in-line measurement for chemical and isotopic analysis.

Operational considerations for each portion of sampling system are given below:

A. Reactor Coolant Sampling System

Sampling of the reactor coolant system is initiated by opening isolation valves (including containment isolation valves using CIAS override) and purging sample flow through the PASS sample vessel/heat exchanger, where it is cooled, through a throttle valve to reduce the pressure, through the in-line chemistry analyses equipment, to the reactor drain tank. At lower pressures, containment sump sample flow is purged in this manner using the HPSI pump discharge mini-flow sample line connection. A float valve downstream of the throttle valve allows for automatic venting of gases coming out of solution. This venting is required to prevent gas bubble interference with flow rate and chemistry measurements in the downstream instrumentation. After sufficient purging, the boron concentration and pH are measured by the in-line instrumentation. A pressurized sample is then collected by isolating the sample vessel/heat exchanger. Total dissolved gas concentration is determined by degassing the sample. This is accomplished by depressurization and circulation by alternate operation of the burette isolation valve and the sample circulation pump. The resulting displacement of liquid into the burette is used to calculate the dissolved gas concentration. The collected gases which have been stripped from the liquid are then directed through a float valve for moisture separation and circulated through a hydrogen and oxygen meter to determine these gas concentrations. Then, the gas sample vessel, which contains nitrogen, is placed online to dilute the gas sample and permit sample withdrawal via a long needle syringe placed through a removable plug in the shield wall and penetrating the rubber wall of the sampler vials for subsequent radioisotope quantification. Additional dilution, which may be necessary for this quantification, may be performed by subsequent nitrogen addition, circulation, and venting. A small fixed volume of depressurized liquid sample (collected in the 4-way valve located in the chemistry analysis line) is then drained to the depressurized sample vessel. This volume is diluted with demineralized water to allow liquid sample withdrawal via syringe for subsequent radioisotope quantification. The radiochemistry and gaseous measurement portions of the system are flushed with demineralized water and purged with N_2 to reduce personnel exposure during withdrawal of the sample and to remove contamination between collection of samples. 23

An undiluted, depressurized and cooled grab sample can be obtained downstream of the PASS station by diverting the liquid sample outlet flow through a shielded sample container located outside the south shield wall between the germanium detector units. The shielded sample container can be disconnected from

the sample system and lifted to the floor above the PASS room
23 for transfer to the station hot laboratory or to an off-site hot
laboratory for chloride analysis or other confirmatory analyses.

B. Containment Atmosphere Sampling System

Sampling of the containment atmosphere under both positive and negative containment pressure conditions is initiated by opening the containment isolation valves by overriding CIAS as necessary and using an air pump to purge the containment air, obtaining H_2 concentration and online isotopic analysis before returning it to the containment. A retained gas sample can be diluted by pumping it through a large vessel containing nitrogen. The diluted sample is then withdrawn via syringe for subsequent radioisotope quantification.

23 An undiluted grab sample of containment air can be obtained
downstream of the PASS station in a manner similar to that used
for the undiluted liquid sample described in paragraph A above.
A separate, removable shielded gas sample container will be
located outside the south shield wall adjacent to the liquid
grab sample cask. The shielded gas grab sample will be removed
from the PASS room via the access hatch for confirmatory
analyses as required either at the station hot laboratory or at
a suitable off-site hot laboratory.

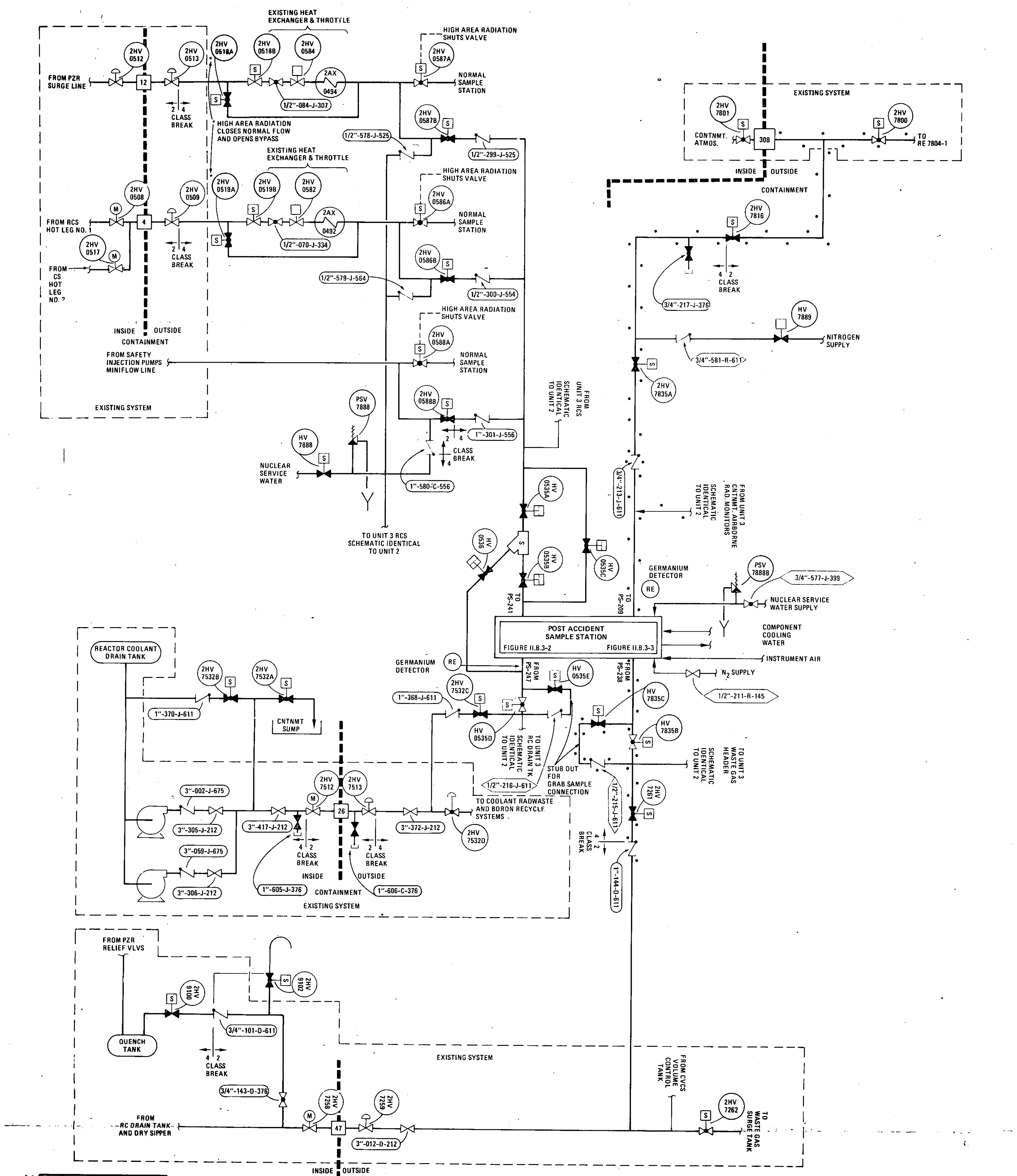
1.4 MEETING NRC REQUIREMENTS

The process sampling system discussed in FSAR subsection 9.3.2, in conjunction with the post-accident sampling system, discussed in subsection 9.3.6 fullfills the NRC requirements as outlined in NUREG-0578, Clarification to NUREG-0578 (NRC letter Nov. 9, 1979), NUREG-0660, NUREG-0694, and NUREG 0737.

23 Procedures for sample collection, transfer, transport and analysis will be
provided in February, 1981. Technical specifications will be provided as
appropriate to supplement the reactor coolant chemistry technical specifica-
tions.

REFERENCE

23 FSAR subsection 9.3.2; and subsection 9.3.6 has been added.



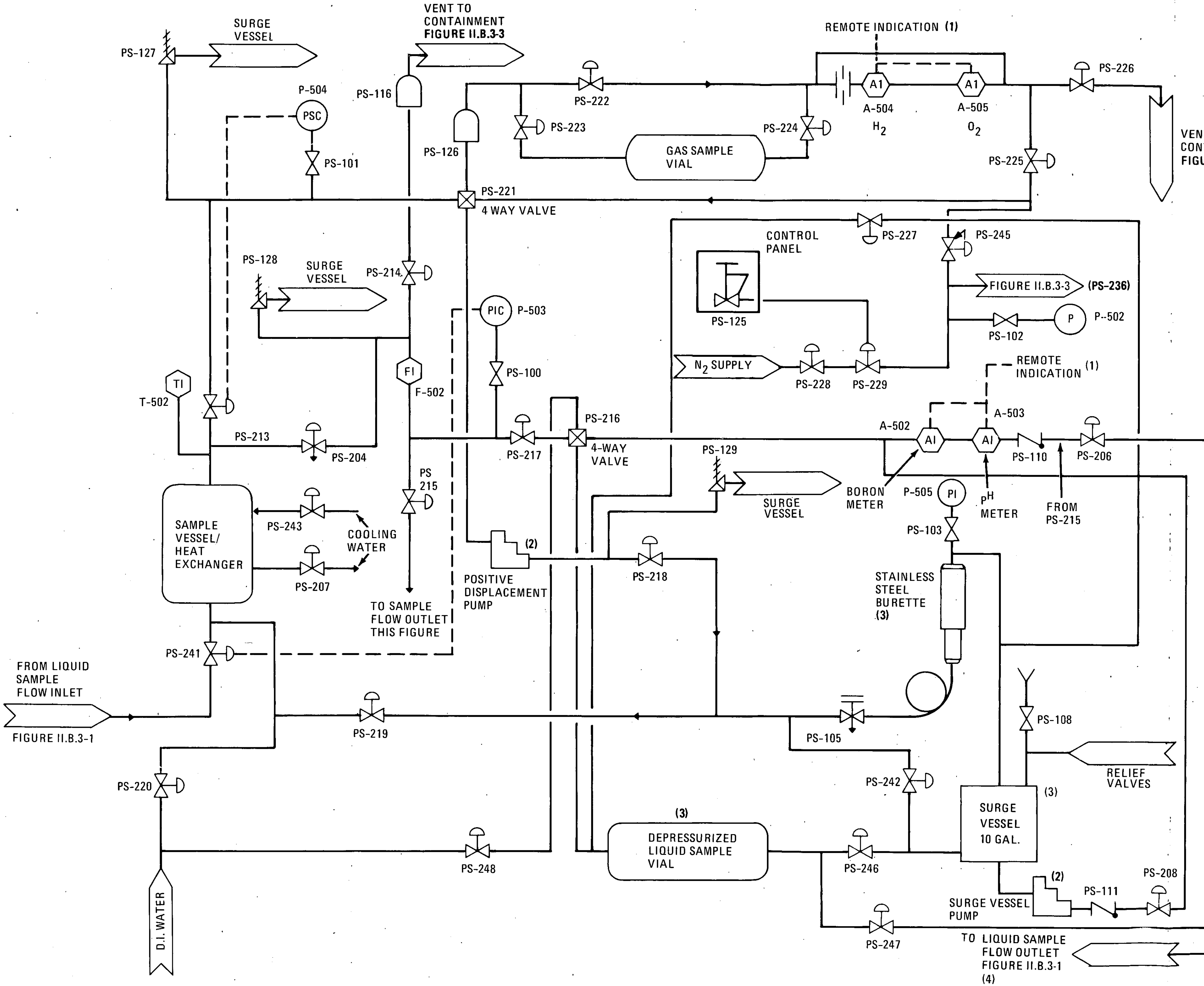
2/81

SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

POST-ACCIDENT SAMPLING SYSTEM

Figure II.B.3-1

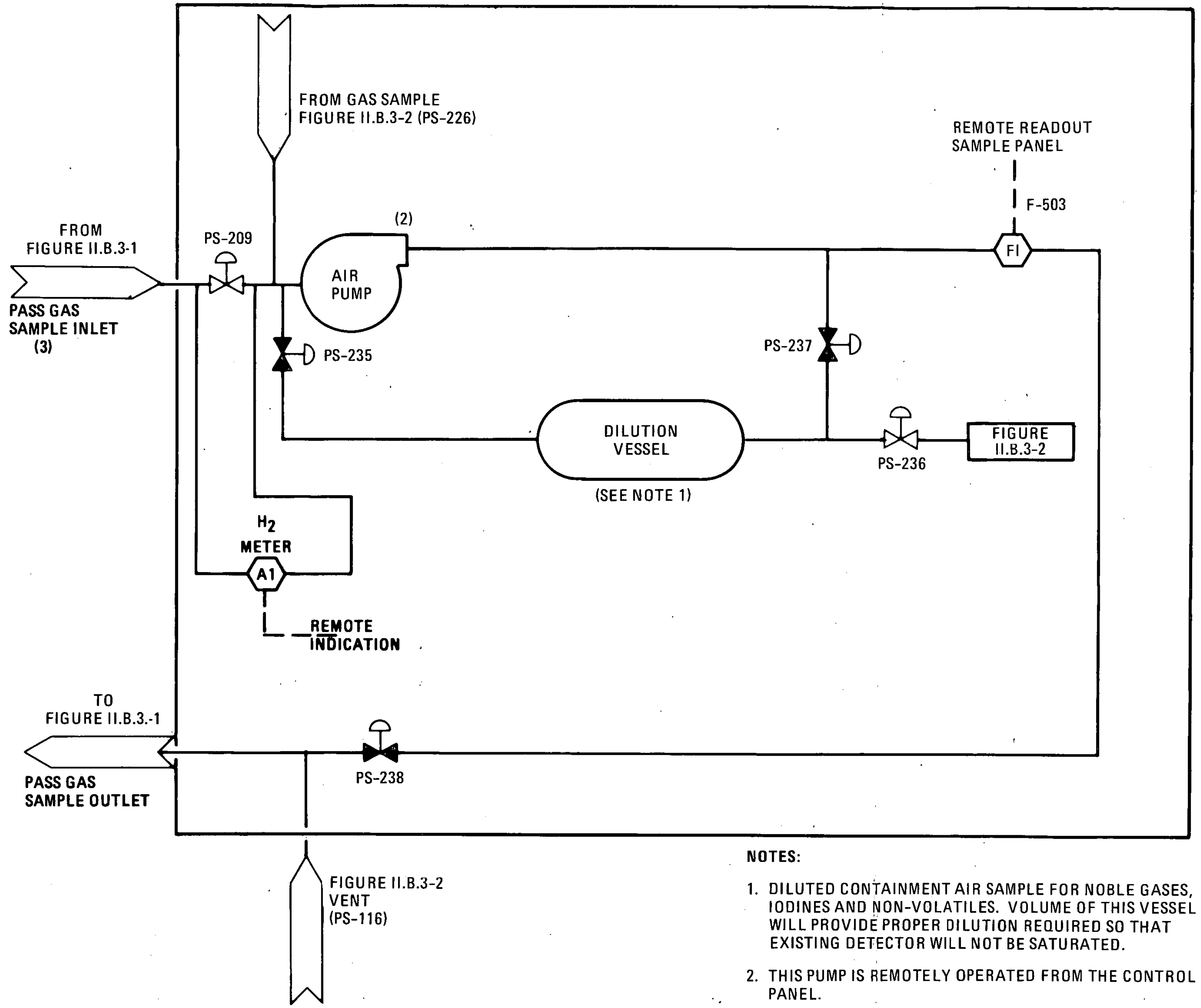
Amendment 23



NOTES:

1. ALL RESULTS WILL BE PRINTED OR READ FROM THE CONTROL PANEL.
2. THE PUMP WILL BE REMOTELY OPERATED FROM THE CONTROL PANEL.
3. LEVEL INDICATORS WITH INDICATION AT THE CONTROL PANEL WILL BE PROVIDED FOR THESE COMPONENTS.
4. ONLINE DETECTOR FOR RADIOCHEMISTRY ANALYSIS IS PROVIDED ON OUTLET LINE.

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
SAMPLE STATION REACTOR COOLANT SAMPLING
Figure II.B.3-2



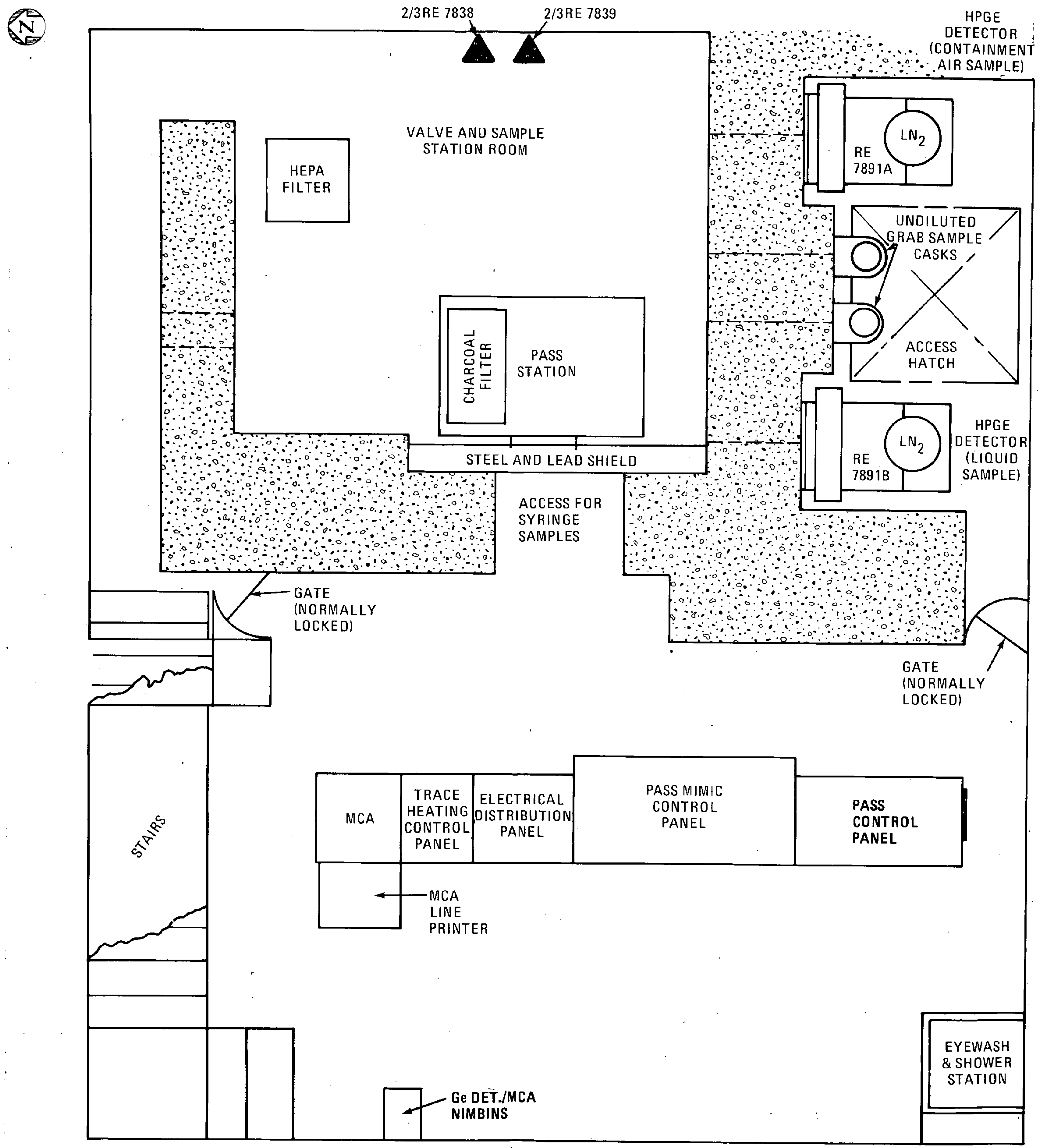
NOTES:

1. DILUTED CONTAINMENT AIR SAMPLE FOR NOBLE GASES, IODINES AND NON-VOLATILES. VOLUME OF THIS VESSEL WILL PROVIDE PROPER DILUTION REQUIRED SO THAT EXISTING DETECTOR WILL NOT BE SATURATED.
2. THIS PUMP IS REMOTELY OPERATED FROM THE CONTROL PANEL.
3. **ON LINE DETECTOR FOR RADIOCHEMISTRY ANALYSIS IS PROVIDED ON INLET LINE.**

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

SAMPLE STATION
CONTAINMENT ATMOSPHERE

Figure II.B.3-3



SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

POST ACCIDENT SAMPLE SYSTEM
ROOM ARRANGEMENT
RAD. WASTE BLDG. 24' LEVEL

Figure II.B.3-4

II.B.4 - NUREG 0737 TRAINING FOR MITIGATING CORE DAMAGE

REQUIREMENT

Position

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

Clarification

Shift technical advisors and operating personnel, from the plant manager through the operations chain to the licensed operators, shall receive all the training indicated in Enclosure 3 to H. R. Denton's March 28, 1980 letter.

Managers and technicians in the Instrumentation and Control (I&C), health physics, and chemistry departments shall receive training commensurate with their responsibilities.

RESPONSE

The station operator training program incorporates the training requirements identified by this position. Station personnel will complete appropriate training prior to full power operation.

The outline of this program is as follows:

A. INCORE INSTRUMENTATION

1. Review the use of incore detectors and thermocouples for providing operator information indicative of core damage.

2. Review other methods for data retrieval such as data from the plant computer.

B. EXCORE NUCLEAR INSTRUMENTATION

1. Review the use of excore detectors in determining the presence of voids.

2. Discuss response of excore detectors in interpreting data.

C. VITAL INSTRUMENTATION

1. Review the design bases of vital instrumentation.

2. Discuss failure modes for vital instrumentation.

3. Review and discuss alternative methods for measurement of vital parameters.

D. PRIMARY CHEMISTRY

1. Review impact on coolant chemistry as consequence of core damage.

2. Discuss operator role in containing accident grade water.

E. RADIATION MONITORING

1. Review and discuss use of area and process radiation monitors following core damage.

2. Review and discuss limitations of radiation monitors.

F. GAS GENERATION

1. Review the sources of gases.

2. Discuss hazardous concentrations of hydrogen and oxygen.

3. Discuss methods for removal of gases.

This program will be presented to the license candidates beginning September, 1980. The training program is discussed in FSAR section 13.2.

REFERENCES

None

II.D.1 - NUREG 0737 - PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2)

REQUIREMENT

Position

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant systems relief and safety valves under expected operating conditions for design-basis transients and accidents.

Clarification

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. 23
Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

A. Performance Testing of Relief and Safety Valves - The following information must be provided in report form by October 1, 1981:

1. Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

2. Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

3. Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

B. Qualification of PWR Block Valves - Although not specifically listed as a short-term lessons-learned requirement in NUREG-0578, qualification of PWR block valves is required by the NRC Task Action Plan NUREG-0660 under task item II.D.1. It is the understanding of the NRC that testing of several commonly used block valve designs is already included in the generic EPRI PWR safety and relief valve testing program to be completed by July 1, 1981. By means of this letter, NRC is establishing July 1, 1982 as the date for verification of block valve functionability. By July 1, 1982, each PWR licensee, for plants so equipped, should provide evidence supported by test that the block or isolation valves between the pressurizer and each power-operated relief valve can be operated, closed, and opened for all fluid conditions expected under operating and accident conditions.

C. ATWS Testing - Although ATWS testing need not be completed by July 1, 1981, the test facility should be designed to accommodate ATWS conditions of approximately 3200 to 3500 (Service Level C pressure limit) psi and 700°F with sufficient capacity to enable testing of relief and safety valves of the size and type used on operating pressurized-water
23 reactors.

RESPONSE

A qualification testing program of reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents has been undertaken on an industry-wide generic basis rather than by individual licensees. The Electric Power Research Institute is conducting the qualification testing program.

By letter dated December 17, 1979, Mr. Williams J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force, submitted "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems", December 13, 1979.

This program plan has since been revised, as reflected in the Revision 1 report dated July 1, 1980. Southern California Edison considers the EPRI Program to be responsive to the above NRC position and accordingly has committed to support and participate in the generic program. A discussion of the program follows.

The EPRI program plan includes tests on full scale PWR safety and relief valves. The program is intended to verify the operational characteristics of PWR safety and relief valves. The objective is to provide adequate assurance that the valves can perform as required to prevent overpressurization of the primary coolant boundary. In addition, the effects of piping upon valve operability will be addressed by obtaining hydraulic and structural piping data from the valve tests. EPRI intends to use this data to validate generic analytical models which can in turn be applied on a plant-unique basis to demonstrate piping adequacy.

The overpressure protection system of San Onofre Units 2&3 includes two ASME Code safety valves located on the pressurizer. The valves are designed to protect the reactor coolant system as required by Section III of the ASME Code.

The San Onofre 2 and 3 designs do not include pressurizer relief valves or block valves.

A description of these safety valves is provided in FSAR subsection 5.4.13 and table 5.4-13. The valves are similar in design to the type of safety valves which will be tested in the EPRI program.

A description of the San Onofre safety valve system was provided to EPRI in January, 1980, and was further updated in November, 1980. Included were safety valve drawings, piping isometric drawings including piping support data, and safety valve operating parameters. This information has been factored into the EPRI program. The EPRI test matrix as outlined in the program plan (Revision 1, July 1, 1980) shows that the test valves will be subjected to conditions and transient simulations bounding those the San Onofre valves are expected to experience. The specific justification of test conditions as well as valve selection applicability will be provided in the final EPRI report. 23

REFERENCES

FSAR subsection 5.4.13 and table 5.4-13. No FSAR changes were made.

II.D.3 - NUREG 0737 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION

REQUIREMENT 23

Position

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

Clarification

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis of an action.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift) and in accordance with Commission Order, May 23rd, 1980 (CLI-20-81).

6. It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration: 23

   a. the use of this information by an operator during both normal and abnormal plant conditions,

   b. integration into emergency procedures,

   c. integration into operator training, and

   d. other alarms during emergency and need for prioritization of alarms.

RESPONSE

San Onofre Units 2 and 3 do not utilize power-operated relief valves, but use safety valves on the pressurizer.

The pressurizer safety valves at San Onofre Units 2 and 3 will be monitored for valve position by an acoustic monitoring system supplied by the Technology for Energy Corporation. The systems (one for each unit) are fully redundant with two channels of instrumentation for each valve to provide positive valve position indication. Each channel consists of an accelerometer (acoustic sensor), a charge converter, and a flow indicator. A flow rate beyond a predetermined setpoint (indicative of a "valve open" position) will cause annunciation on the main control panel in the control room with bar graph indication of relative flow rate (range scales from 0 to 1.0; 1.0 being full flow).

Module calibration is done at the factory and normally does not require recalibration. System calibration, as well as lower and upper alarm setpoint adjustment, is completed in the field since this calibration is dependent on individual system background noise levels and sensor location.

The system will be safety grade and seismically qualified; seismic Category I, Quality Class II. The system will be qualified to function for its appropriate environment (including accidents that would cause the safety valves to lift) for both equipment inside and outside of containment. The system will be installed prior to fuel load and the qualification program for this system will be completed prior to operation above 5% power.

A human factors analysis of the control room is being conducted as part of
23 the control room design review discussed in item I.D.1.

REFERENCES

None.

II.E.1.1 - NUREG 0737 AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

REQUIREMENT

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

1. Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

2. Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

3. Reevaluate the AFW system flowrate design bases and criteria.

Clarification 23

Operating Plant Licenses; Item 1 and 2 above, have been completed for Westinghouse (W), Combustion Engineering (C-E), and two Babcock and Wilcox (B&W) operating plants (Rancho Seco, short-term only, and TMI-1). As a result of staff review of items 1 and 2, letters were issued to these plants that required the implementation of certain short and long term AFW system upgrade requirements. Included in these letters was a request for additional information regarding item 3 above. The staff is now in the process of evaluating licensees' responses and commitments to these letters.

The remaining B&W operating plants (Oconee 1-3, Crystal River 3, ANO-1, and Davis-Besse 1) have submitted the analysis described in item 1 above. The analysis is presently undergoing staff review. When the results of the staff reviews are complete, each of the remaining B&W plants will receive a letter specifying the short and long term AFW system upgrade requirements based on item 1 above. Included in these letters will be a request for additional information regarding items 2 and 3 above.

Operating License Applicants; Operating license applicants have been requested to respond to staff letters of March 10, 1980 (W and C-E) and April 24, 1980 (B&W). These responses will be reviewed during the normal review process for these applications.

RESPONSE

The auxiliary feedwater system (AFWS) design for San Onofre Units 2&3 is being modified in response to NRC concerns regarding the system's ability to meet AFWS pipe break criteria. The design of the AFWS is described in FSAR subsection 10.4.9. A third 100% capacity motor-driven pump has been added to the system as shown on the simplified process flow diagram (figure II.E.1.1-1). The modified design has the following features, which are discussed in the AFWS reliability evaluation.

- A third motor-driven pump will be installed.

- A third pump will be powered from the diesel separate from the existing motor pump. In addition, automatic actuation will be utilized (see section II.E.1.2, AFWS Automatic Initiation and Flow Indication).

- For certain breaks in the AFWS, operator action is required to isolate the break.

An outline of the evaluation planned in response to the NRC's March 10, 1980 letter to all pending operator license applicants regarding the AFWS was presented to the NRC in a meeting of May 15, 1980. The evaluation was pro-
23 vided to NRC by SCE letter dated November 24, 1980. This evaluation was presented using the San Onofre system as an example. Substantive points covered in the evaluation outline were as follows:

- Deterministic Review

- Reliability Evaluation

- Design Bases Review

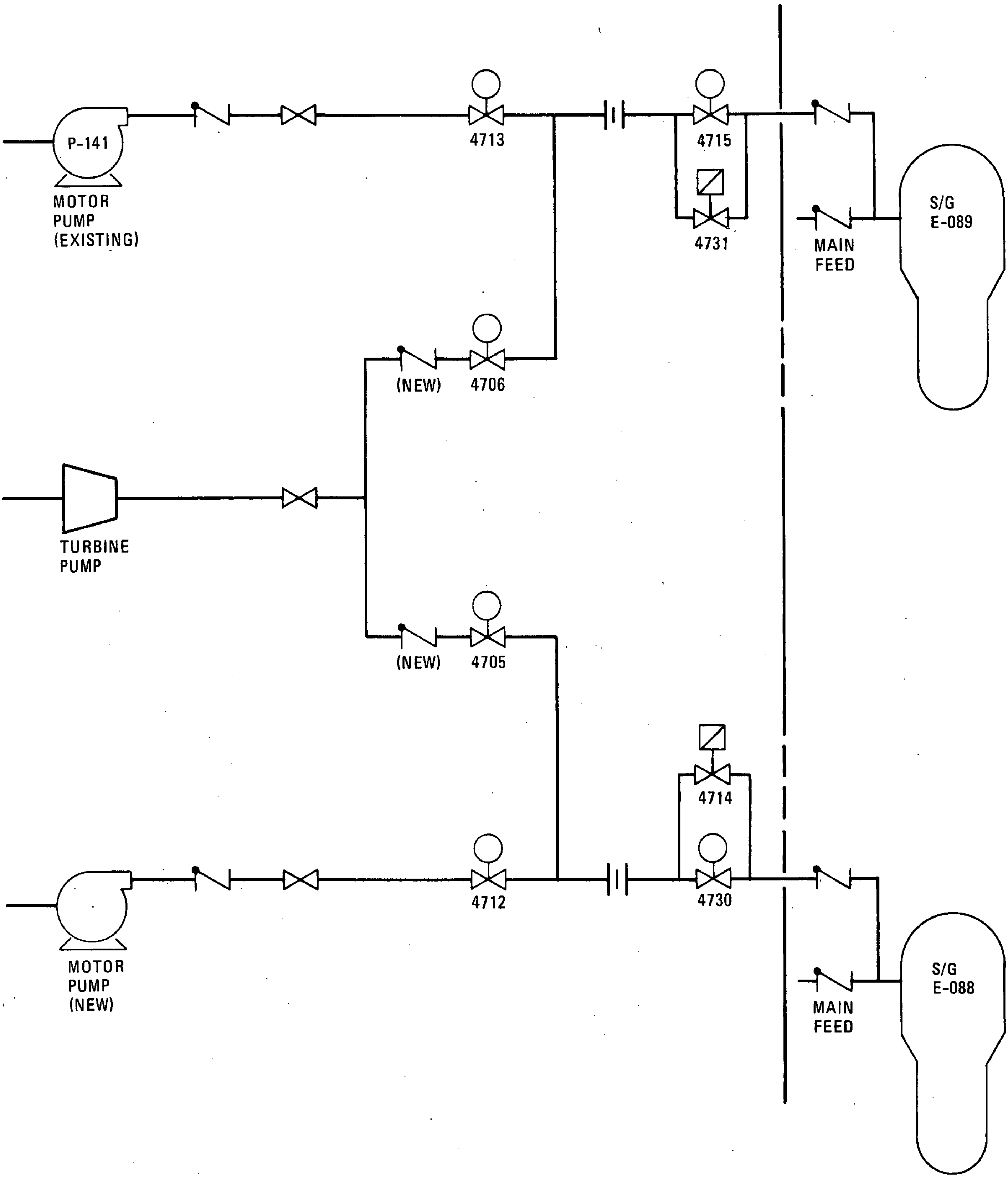
- Addressing Operating Plant Requirements

The full implementation and system design interpretation of the Staff guidance for the AFWS stated in SRP 10.4.9 is presented to the NRC in the evaluation discussed above and in a FSAR subsection 10.4.9.

All changes to the AFWS will be completed prior to 100% power operation.

REFERENCES

See revised FSAR subsection 10.4.9.



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

AFWS DESIGN

Figure II.E.1.1-1

II.E.1.2 NUREG 0737 AUXILIARY FEEDWATER INITIATION AND INDICATION

REQUIREMENT

PART 1: Auxiliary Feedwater System Automatic Initiation

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system (AFWS), the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the AFWS.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses. 23

5. Manual capability to initiate the AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

Clarification

The intent of this recommendation is to assure a reliable automatic initiation system. This objective can be met by providing a system which meets all the requirements of IEEE Standard 279-1971.

The staff has determined that the following salient paragraphs of IEEE 279-1971 should be addressed as a minimum:

IEEE 279-1971, Paragraph	
4.1	General Functional Requirements
4.2	Single Failure
4.3, & 4.4	Qualification
4.6	Channel Independence
4.7	Control and Protection System Interaction
4.9 & 4.10	Capability for Testing
4.11	Channel Bypass
4.12	Operating Bypass
4.13	Indication of Bypass
4.17	Manual Initiation

PART 2: Auxiliary Feedwater System Flowrate Indication

Position

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain the
23 actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Clarification

The intent of this recommendation is to assure a reliable indication of AFWS performance. This objective can be met by providing an overall indication system that meets the following appropriate design principles:

1. For Babcock and Wilcox Plants

   a. To satisfy these requirements, B&W plants must provide as a minimum two auxiliary feedwater flowrate indicators for each steam generator.

b. The flow indication system should conform to the following salient paragraphs of IEEE 279-1971:

IEEE 279-1971, Paragraph	
4.1	General Functional Requirements
4.2	Single Failure
4.3 & 4.4	Qualification
4.6	Channel Independence
4.7	Control and Protection System Interaction
4.9 & 4.10	Capability for Testing

2. For Westinghouse and Combustion Engineering Plants

a. To satisfy these requirements, W and C-E plants must provide as a minimum one auxiliary feedwater flowrate indicator and one wide-range steam-generator level indicator for each steam generator or two flowrate indicators.

b. The flow indication system should be: 23

(1) Environmentally qualified
(2) Powered from highly reliable, battery-backed non-Class IE power source
(3) Periodically testable
(4) Part of plant quality assurance program
(5) Capable of display on demand

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

- The use of this information by an operator during both normal and abnormal plant conditions,
- Integration into emergency procedures,
- Integration into operator training, and
- Other alarms during emergency and need for prioritization of alarms.

RESPONSE

The auxiliary feedwater system for San Onofre Units 2 and 3 is described in FSAR subsection 10.4.9. The AFWS design is being modified to respond to NRC concerns regarding AFWS pipe break criteria. A third 100% capacity motordriven pump has been added to the system. This pump and its associated valves are automatically operated using the same logic criteria as the existing system. A discussion of the changes required to assure that the modified system continues to comply with the above NRC position is provided

23 in the response to NUREG 0660/0737 item II.E.1.1. Addition of the third pump has not changed the logic criteria for system initiation nor has it necessitated any changes to the existing flow indication. All changes to the AFWS will be completed prior to 100% power operation.

23 PART 1: Initiation

1. The AFWS initiating circuitry incorporates both automatic and manual system start capability. The automatic initiation of the auxiliary feedwater system is described in FSAR paragraph 7.3.1.1.7. The parameters monitored are steam generator level and pressure.

2. The automatic initiation signals and circuits are designed so that a single failure will not result in the loss of auxiliary feedwater system function as described in FSAR paragraph 7.3.1.1.7.

3. Testability of the initiating signals and circuits is described in FSAR paragraph 7.3.1.1.7.

4. The initiating signals and circuits are powered from the 1E emergency power supplies.

5. Manual initiation of the auxiliary feedwater system is from main control room panel CR52&53. A single failure in the manual circuits will not result in the loss of system function as described in FSAR paragraph 7.3.1.1.7.

6. The ac motordriven auxiliary feedwater pumps and valves are automatically actuated and sequenced to the emergency bus as described in FSAR table 8.31.

   23 Additional evidence of full compliance with applicable criteria of IEEE 279-1971 can be found in FSAR section 7.3.2.1.2, paragraphs A through V and in section 7.3.2.1.3.

7. The automatic initiation signals and circuits are designed so that their failure will not result in the loss of manual capability to initiate the auxiliary feedwater system from the control room.

23 PART 2: Indication

1. The existing auxiliary feedwater system design for San Onofre Units 2 and 3 provides one channel of safetygrade flow indication per steam generator. See FSAR section 7.5 for a discussion of safetyrelated display instrumentation. Each channel provides an indication of feedwater flow within an accuracy of $\pm$ 10%. The instrumentation channels are powered from the safety grade instrument buses. This preferred power source is considered to be consistent with the intent of the emergency power diversity requirements of NRC Branch Technical Position 10-1 in that no failure in preferred offsite or backup onsite sources (diesel generators) will cause a common mode loss of all auxiliary feedwater flow indication.

2. To satisfy single failure criteria, a backup means for verifying auxiliary feedwater flow is provided by the existing four channels of narrow range safetygrade level indication per steam generator plus two new channels of wide range safetygrade steam generator level indication per steam generator. The narrow range and wide range channels are all powered from safety grade instrument buses. See FSAR section 7.5 for a discussion of safetyrelated display instrumentation. 23

3. A human factors analysis of the control room is being conducted as part of the control room design review discussed in item I.D.1. Auxiliary Feedwater flow instrumentation is included in that review.

4. Electrical schematics of the AFW flow and wide range steam generator level instrument loops have been provided as documented in FSAR table 1.7-1 (drawing numbers 504-3-341, 504-3-355, 944-580, and 944-581). There are no associated logic diagrams. The AFW P&ID is FSAR figure 10.4-9. Steam generator level indication is shown on FSAR figure 10.1-1, sheet 1. 23

   Test procedures will be developed as part of the normal startup test program and will be available for NRC staff review.

   Technical Specification 3/4.3.3.6 will be revised to include AFW Flow and Steam generator wide range level as accident monitoring instrumentation.

REFERENCE

FSAR section 7.5, subsection 10.4.9, and 7.3.1.1.7, 7.3.2.1.2, 7.3.2.1.3, and table 8.3-1, and NUREG 0660, Item II.E.1.1. 23

II.E.3.1 - NUREG 0737 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

REQUIREMENT 23

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Clarification

1. Redundant heater capacity must be provided, and each redundant heater or group of heaters should have access to only one Class 1E division power supply. 23

2. The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

3. The power sources need not necessarily have the capacity to provide power to the heaters concurrent with the loads required for LOCA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   a. Which ESF loads may be appropriately shed for a given situation.

   b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.

   c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Reg. Guide 1.75)

7. Being non-Class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. (See item 5.b. above)

RESPONSE

General Description

The pressurizer electrical heater system consists of two banks of proportional heaters and six banks of back-up heaters. The proportional heaters are used to correct small pressure variations and are powered from non safety-related (non-emergency) buses. The back-up heaters are used to correct larger pressure variations. Of the six back-up heater banks, four are powered from a non safety-related bus. The controls and motive power to each of two remaining backup heater banks are from separate safety-related (emergency) buses under all conditions.

Design Criteria

Either of the two class IE powered backup heater banks are adequate to maintain reactor coolant system pressure control during a natural circulation cooldown on a complete loss of offsite power. There are three modes of operation of these two heater banks i.e., AUTO, MANUAL, and OVERRIDE. The following is a description of the three modes of operation.

A. Auto Mode

   In this mode of operation the heaters are automatically turned on or off depending on the pressurizer pressure and level.

B. Manual Mode

   In the manual mode of operation, the non safety-related controls associated with the pressurizer level and pressurizer control

II.E.3.1 - NUREG 0737 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

REQUIREMENT 23

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Clarification

1. Redundant heater capacity must be provided, and each redundant heater or group of heaters should have access to only one Class 1E division power supply. 23

2. The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

3. The power sources need not necessarily have the capacity to provide power to the heaters concurrent with the loads required for LOCA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

    a. Which ESF loads may be appropriately shed for a given situation.

    b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.

    c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Reg. Guide 1.75)

7. Being non-Class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. (See item 5.b. above)

RESPONSE

General Description

The pressurizer electrical heater system consists of two banks of proportional heaters and six banks of back-up heaters. The proportional heaters are used to correct small pressure variations and are powered from non safety-related (non-emergency) buses. The back-up heaters are used to correct larger pressure variations. Of the six back-up heater banks, four are powered from a non safety-related bus. The controls and motive power to each of two remaining backup heater banks are from separate safety-related (emergency) buses under all conditions.

Design Criteria

Either of the two class IE powered backup heater banks are adequate to maintain reactor coolant system pressure control during a natural circulation cooldown on a complete loss of offsite power. There are three modes of operation of these two heater banks i.e., AUTO, MANUAL, and OVERRIDE. The following is a description of the three modes of operation.

A. Auto Mode

In this mode of operation the heaters are automatically turned on or off depending on the pressurizer pressure and level.

B. Manual Mode

In the manual mode of operation, the non safety-related controls associated with the pressurizer level and pressurizer control system are blocked, and in the event of a loss of offsite power the operator can take manual control of the heaters. The heater can be put in a manual mode by operating the "ON" or "OFF" position pushbutton of the control switch.

C. Override Mode

Since the pressurizer heater elements are non safety grade, the two banks powered from emergency buses are automatically isolated by a safety injection or emergency feedwater actuation signals. Power to the heaters is automatically disconnected to ensure that the safety grade power systems required for ECCS and secondary cooling systems are not degraded consistent with the separation criteria in FSAR section 8.1.4. However an override switch is provided to enable the operator to override these automatic signals at the component level and reconnect the breakers should it be necessary to use the pressurizer heaters. Use of this override feature is administratively controlled. Consistent with override control for other ESF systems, override is possible only when the safety signal is present and the override condition is annunciated in the control room. To prevent heater damage, all heaters are tripped on low-low pressurizer level. The level control is safety-related. Procedures and training will be provided to make operators aware of when and how the required pressurizer heaters shall be reconnected to the 23
emergency buses.

MEETING NRC REQUIREMENT

FSAR subsection 8.1.4 describes the onsite and offsite power systems with 23
the appropriate design criteria, regulatory guides, and IEEE standards. The implementation of the current write-up together with FSAR subsection 8.1.4 satisfies the NRC requirements outlined in NUREG 0578, Clarification to NUREG 0578 (NRC Letter, November 9, 1979), NUREG 0660, and NUREG 0737. 23

Adequate heater capacity, capable of controlling RCS pressure during a natural circulation cooldown following loss of offsite power, is provided by either of the two banks of backup heaters. Operator action is required to take heaters out of the automatic mode and/or override the automatic tripping of heaters in SIAS or EFAS.

DOCUMENTATION

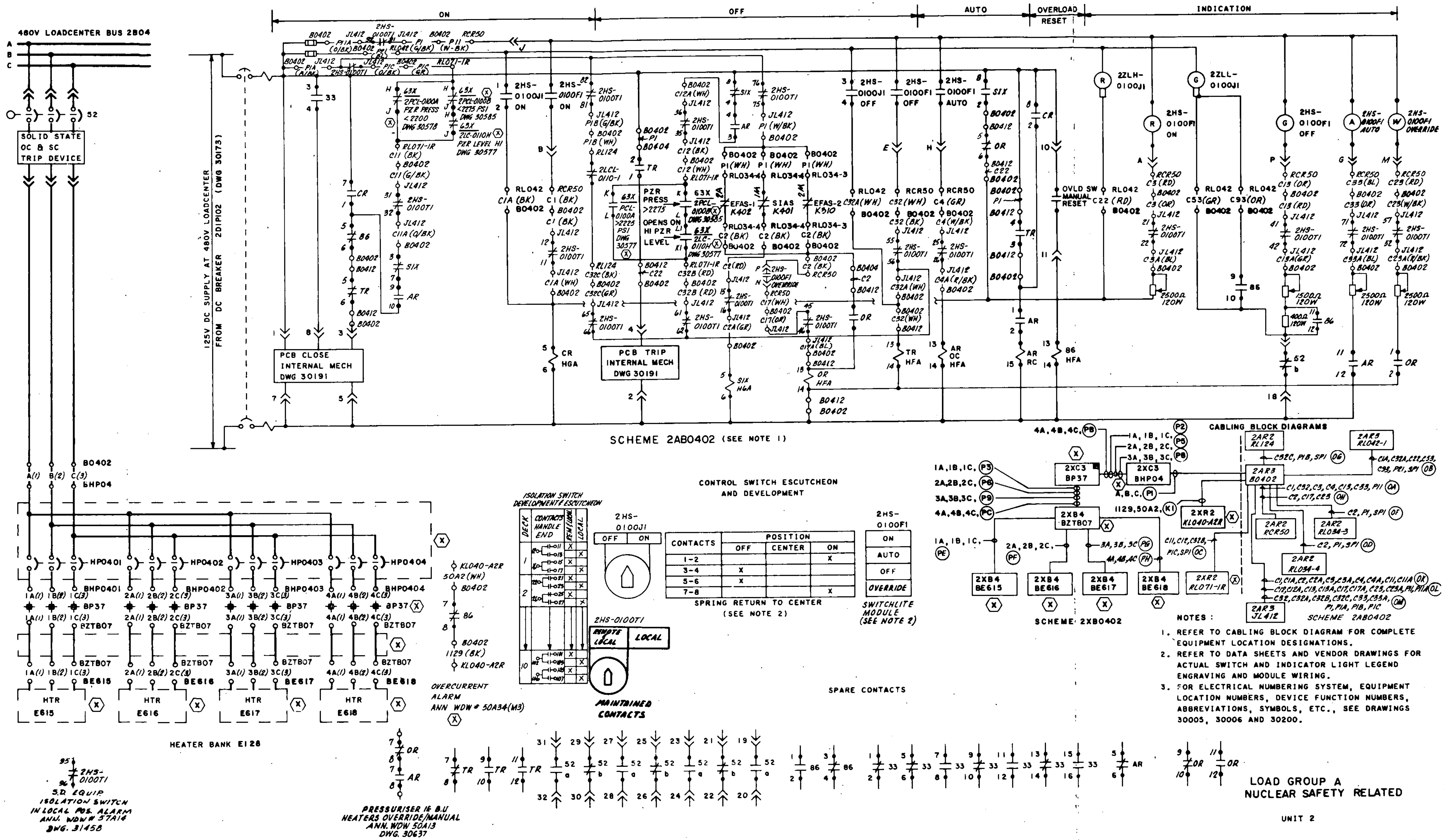
Electrical schematics of the pressurizer heater groups supplied by the safety-related buses are shown in figures II.E.3.1-1 and II.E.3.1-2. 23
There are no associated control logic diagrams.

Test procedures will be developed as part of the normal startup test program and will be available for NRC staff review.

23 Technical Specification 3/4.4.4 will include operability requirements for the pressurizer heaters supplied from the emergency buses.

REFERENCE

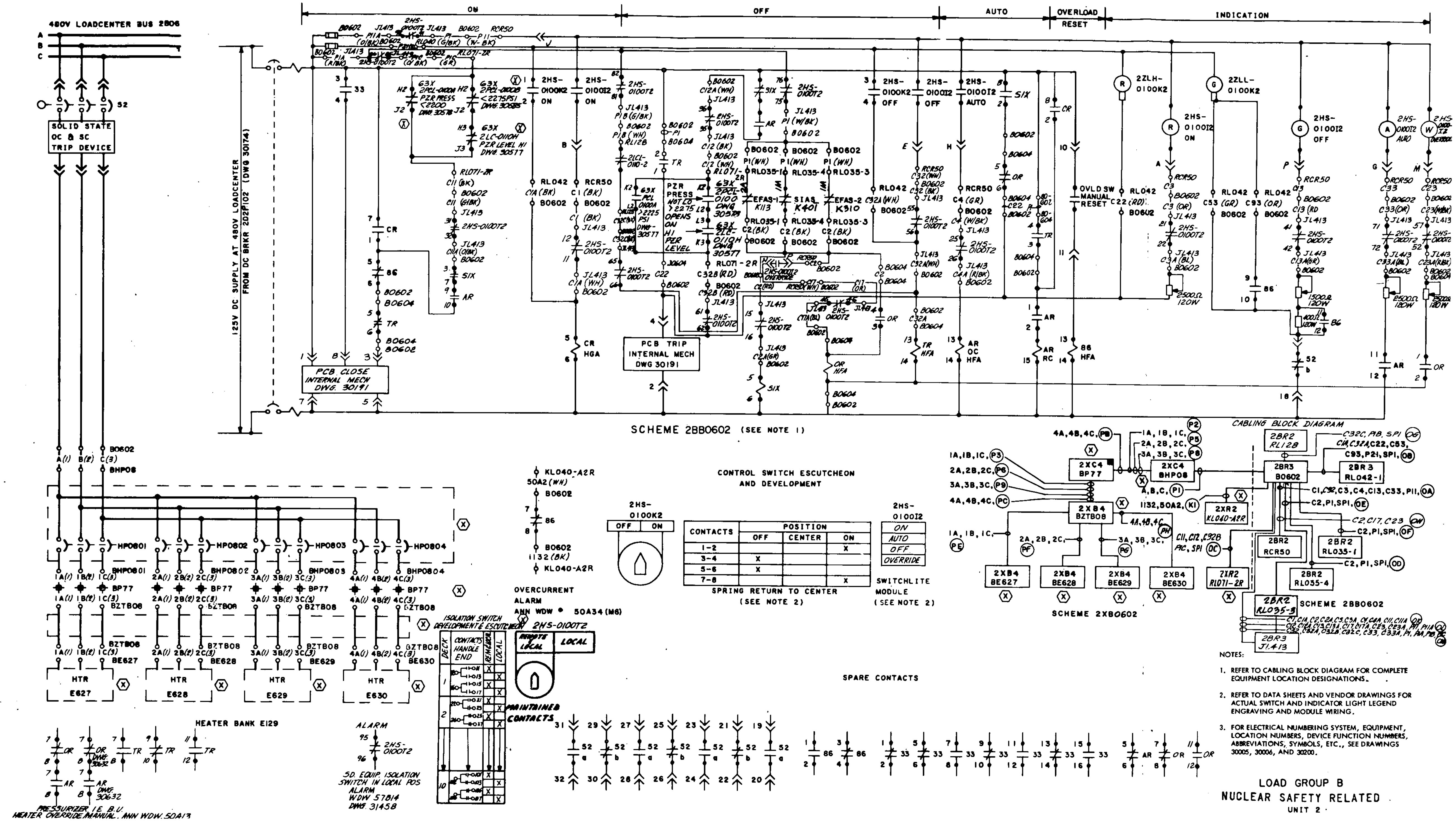
FSAR subsection 8.1.4. No FSAR changes were made.



SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

ELEMENTARY DIAGRAM
REACTOR - PRESSURIZER
BACKUP HTRS BANK E 128

Figure II.E.3.1-1



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

ELEMENTARY DIAGRAM
REACTOR - PRESSURIZER
BACKUP HTRS BANK E 129

Figure II.E.3.1-2

II.E.4.1 - NUREG 0737 DEDICATED HYDROGEN PENETRATIONS

REQUIREMENT

Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment penetration sytems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

Clarification

1. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system. 23

2. The dedicated penetration or the combined single-failure proof alternative shall be sized such that the flow requirements for the use of the recombiner or purge system are satisfied. The design shall be based on 10 CFT 50.44 requirements.

3. Components furnished to satisfy this requirements shall be safety grade.

4. Licensees that rely on purge systems as the primary means of controlling combustible gases following a loss-of-coolant accident should be aware of the positions taken in SECY-80-399, "Proposed Interim Amendments to 10 CFR Part 50 Related to Hydrogen Control and Certain Degraded Core Considerations." This proposed rule, published in the Federal Register on October 2, 1980, would require plants that do not have recombiners to have the capacity to install external recombiners by January 1, 1982. (Installed internal recombiners are an acceptable alternative to the above.)

5. Containment atmosphere dilution (CAD) systems are considered to be purge systems for the purpose of implementing the requirements of this TMI Task Action item.

RESPONSE

The San Onofre Units 2&3 combustible gas control systems were reviewed, according to NUREG 0737 Action Item II.E.4.1, and no revisions were deemed necessary. 23

The hydrogen recombiners are located within the containment structure, on the operating deck, well above the maximum containment post-LOCA water level. There are two completely separate and redundant recombiners per unit, powered from separate Class IE electrical buses. Thus, a single failure will not prevent the recombiners from performing their safety function. Also, since the recombiners are located inside the containment structure, operating personnel are shielded from the recombiners preventing any radiation exposure.

Thus, San Onofre Units 2 and 3 fully comply with all NRC positions
23 presented in NUREG 0578, NRC Letter, November 9, 1979, NUREG 0660, and NUREG 0737.

REFERENCE

FSAR subsection 6.2.5, Combustible Gas Control in Containment. No FSAR changes were made.

II.E.4.2 - NUREG 0737 CONTAINMENT ISOLATION DEPENDABILITY

REQUIREMENT 23

Position

1. Containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plant personnel shall give careful reconsideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to NRC. 23

3. All nonessential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

5. The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

6. Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days. 23

7. Containment purge and vent isolation valves must close on a high radiation signal.

Clarification

1. The reference to SRP 6.2.4 in position 1 is only to the diversity requirements set forth in that document.

2. For post-accident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57, as clarified by Standard Review Plant, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration must receive the diverse isolation signals.

3. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus nonessential systems and is due to be issued by June 1981. Requirements for operating plants to review their list of essential and nonessential systems will be issued in conjunction with this guide including an appropriate time schedule for completion.

4. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting position 4.

23 5. Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

6. The containment pressure history during normal operation should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 lb./in.^{2} above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 lb./in.^{2} will require detailed justification. Applicants for an operating license and operating plant licensees that have operated less than one year should use pressure history data from similar plants that have operated more than 1 year, if possible, to arrive at a minimum containment setpoint pressure.

7. Sealed-closed purge isolation valves shall be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

RESPONSE

The responses to the above NRC positions are as follows:

1. The containment isolation actuation signal (CIAS) is being modified to include diversity in the parameters sensed for initiation as recommended by SRP 6.2.4. CIAS currently takes place on high containment pressure. The modification uses the safety injection actuation signal (SIAS) as the diverse method of providing containment isolation. This will, in effect, utilize low pressurizer pressure as the diverse parameter.

   The design change is being implemented by wiring modifications in the termination cabinets of the ESF auxiliary relay cabinets to provide for actuation of all CIAS components, with the exception of MSIVs and MFIVs, on either high containment pressure or SIAS.

2. An evaluation has been performed to identify the essential and nonessential systems with respect to the capability to override isolation following CIAS. The selection for each essential system for CIAS override capability was based on an engineering evaluation of systems that 23
   are necessary or potentially useful following an accident. Override switches are being provided as shown in table II.E.4.2-1 to allow the operator to manually bring the essential in-containment systems back into service. The conditions for overriding and reinitiating these systems will be specified in the operating procedures.

3. At the present time, both essential and nonessential systems are isolated by the containment isolation signals. These systems are listed in table 6.2-30 of the FSAR. As a result of the evaluation to define the essential and nonessential systems described in (2) above, manual override capabilities are being provided for selected components. The override capabilities are being provided on a selected group basis, and will reinitiate only a specific system. The override switch is a 23
   permissive only and does not open the associated isolation valve(s). The override switch design is consistent with the criteria for safety-related equipment. A list of equipment provided with override capability is given in table II.E.4.2-1.

4. All valves actuated by the CIAS remain closed following recovery from CIAS conditions. The design of the control systems for the valves is such that resetting of the CIAS does not result in the automatic reopening of the valves. Deliberate operator action is required to reopen the valves.

5. The containment isolation actuation system setpoint is 4 lb./in.^{2}g and is considered to be the minimum compatible with normal operation. The channel accuracy assumed in the safety analysis is ±1.5 lb./in.^{2} (refer to FSAR table 7.3-3). Allowing a 1 lb./in.^{2} margin results in 23
   a normal operation maximum pressure of 1.5 lb./in.^{2}g. While SONGS 2/3 has no operating history, this value is generally consistent with operating pressures observed at other facilities with similar containment designs.

6. The containment mini-purge supply and exhaust isolation valves conform to the operability criteria of Branch Technical Position CSB 6-4 (refer to response to Staff question 022.12).

   The containment large volume purge supply and exhaust valves will be 23 sealed closed by administrative control which will consist of checking the valve position lights in the control room once per 24 hours. This requirement will be included in the Technical Specifications.

7. Both the mini-purge and large volume purge supply and exhaust valves are automatically closed on high radiation by the Containment Purge Isolation Signal (refer to FSAR paragraph 7.3.1.1.5 for a description of the CPIS).

All of the above stated changes will be fully implemented prior to fuel load.

In addition, San Onofre Units 2&3 will have operating procedures in place prior to fuel load which will require valve position verification for containment isolation valves discussed above.

REFERENCES

FSAR table 6.2-30:

23

FSAR paragraphs 7.3.1.1, 7.3.1.1.4, 7.3.1.1.4.3, 7.3.1.1.4.9; tables 7.3-2, 7.3-15, 7.3-16; and figures 7.3-1 and 7.3-2 have been revised.

Table II.E.4.2-1
LIST OF VALVES WITH OVERRIDE CAPABILITY (Sheet 1 of 3)

OVERRIDE SWITCH	VALVE	SYSTEM/FUNCTION
HS-9093-2	2HV7911	Nuclear Service Water
HS-9092-1	2HV5803	Containment Sump Pump
HS-9092-2	2HV5804	Containment Sump Pump
HS-5686D2	2HV5686	Fire Protection
HS-4058-1	2HV4058	Steam Generator Secondary Steam Sample
HS-4057-2	2HV4057	Steam Generator Secondary Steam Sample
HS-5388-1	2HV5388	Instrument Air
HS-6397-2	2HV6211	Component Cooling Water to and from RCP's
HS-6397-2	2HV6216	Component Cooling Water to and from RCP's
HS-6397-2	2HV6213	Non-Crit CCW Supply Isolation Valve
HS-6397-2	2HV6219	Non-Crit CCW Return Isolation Valve
HS-6223*	2HV6223	Component Cooling Water to and from RCP's
HS-6236*	2HV6236	Component Cooling Water to and from RCP's
HS-6397-1	2HV6212	Non-Crit CCW Supply Isolation Valve
HS-6397-1	2HV6218	Non-Crit CCW Return Isolation Valve
HS-6374-1**	2HV6366	CCW to and from Containment Em. Coolers
HS-6374-1**	2HV6367	CCW to and from Containment Em. Coolers
HS-6374-1**	2HV6370	CCW to and from Containment Em. Coolers
HS-6374-1**	2HV6371	CCW to and from Containment Em. Coolers

23

*These valves receive no isolation signal, but their operation is necessary to initiate flow to the RCP's. The valves can be remote-manually operated from the control room.

**These values open on CCAS. Override to close is necessary to reduce CCW loads to permit reinitiation of non-critical CCW.

Table II.E.4.2-1
LIST OF VALVES WITH OVERRIDE CAPABILITY (Sheet 2 of 3)

23

OVERRIDE SWITCH	VALVE	SYSTEM/FUNCTION
HS-6374-2**	2HV6368	CCW to and from Containment Em. Coolers
HS-6374-2**	2HV6369	CCW to and from Containment Em. Coolers
HS-6374-2**	2HV6372	CCW to and from Containment Em. Coolers
HS-6374-2**	2HV6373	CCW to and from Containment Em. Coolers
HS-9090-1	2HV9920	Normal Coolers Chilled Water Supply
HS-9090-1	2HV9921	Normal Coolers Chilled Water Return
HS-9090-2	2HV9900	Normal Coolers Chilled Water Supply
HS-9090-2	2HV9971	Normal Coolers Chilled Water Return
HS-9095-1	2TV0221	Letdown
HS-9095-1	2HV9205	Letdown
HS-9095-2	2HV9204	Letdown
HS-9095-2	2TV9267	Letdown
HS-0590-2	2HV0508	Sampling System
HS-0590-2	2HV0510	Sampling System
HS-0590-2	2HV0512	Sampling System
HS-0590-2	2HV0515	Sampling System
HS-0590-2	2HV0517	Sampling System
HS-0590-1	2HV0509	Sampling System
HS-0590-1	2HV0511	Sampling System
HS-0590-1	2HV0513	Sampling System
HS-0590-1	2HV0514	Sampling System

Table II.E.4.2-1
LIST OF VALVES WITH OVERRIDE CAPABILITY (Sheet 3 of 3)

OVERRIDE SWITCH	VALVE	SYSTEM/FUNCTION
HS-7861-1	2HV7802	Containment Airborne Radiation Monitoring System
HS-7861-1	2HV7801	Containment Airborne Radiation Monitoring System
HS-7861-1	2HV7810	Containment Airborne Radiation Monitoring System
HS-7861-1	2HV7816	Containment Airborne Radiation Monitoring System
HS-7861-1	2HV7811	Containment Airborne Radiation Monitoring System
HS-7861-2	2HV7803	Containment Airborne Radiation Monitoring System
HS-7861-2	2HV7805	Containment Airborne Radiation Monitoring System
HS-7861-2	2HV7806	Containment Airborne Radiation Monitoring System
HS-7861-2	2HV7800	Containment Airborne Radiation Monitoring System
HS-9096-2	2HV9217	Reactor Coolant Pump Bleedoff
HS-9096-1	2HV9218	Reactor Coolant Pump Bleedoff
HS-0590-2	2HV7258	Containment Waste Gas Header
HS-0590-1	2HV7259	Containment Waste Gas Header
HS-0590-1	2HV7512	Reactor Coolant Drain Tank to Coolant Radwaste
HS-0590-2	2HV7513	Reactor Coolant Drain Tank to Coolant Radwaste

23

II.F.1 NUREG 0737 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

REQUIREMENT

Introduction

Item II.F.1 of NUREG-0660 contains the following subparts:

1. Noble gas effluent radiological monitor;
2. Provisions for continuous sampling of plant effluents for post-accident releases of radioactive iodines and particulates and onsite laboratory capabilities (this requirement was inadvertently omitted from NUREG-0660; see Attachment 2 that follows, for position);
3. Containment high-range radiation monitor;
4. Containment pressure monitor;
5. Containment water level monitor; and
6. Containment hydrogen concentration monitor.

NUREG-0578 provided the basic requirements associated with items 1. through 3. above. Letters issued to all operating nuclear power plants dated September 13, 1979 and October 30, 1979 provided clarification of staff requirements associated with items 1. through 6. above. Attachments 1 through 6 present the NRC position on these matters. 23

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

a. the use of this information by an operator during both normal and abnormal plant conditions,

b. integration into emergency procedures,

c. integration into operator training, and

d. other alarms during emergency and need for prioritization of alarms.

ATTACHMENT 1 NOBLE GAS EFFLUENT MONITOR

Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

1. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

2. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

Clarification

1. Licensees shall provide continuous monitoring of high-level, post-accident releases of radioactive noble gases from the plant. Gaseous effluent monitors shall meet the requirements specified in the enclosed Table II.F.1-1. Typical plant effluent pathways to be monitored are also given in the table.

2. The monitors shall be capable of functioning both during and following
23 an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

3. Offline monitors are not required for the PWR secondary side main steam safety valve and dump valve discharge lines. For this application, externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required.

4. Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

   The design description shall include the following information.

   a. System description, including:

      (1) instrumentation to be used, including range or sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;

      (2) monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background correction;

(3) location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

(4) assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and,

(5) the source of power to be used.

b. Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

TABLE II.F.1-1

HIGH-RANGE NOBLE GAS EFFLUENT MONITORS

REQUIREMENT - Capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored. 23

PURPOSE - To provide the plant operator and emergency planning agencies with information on plant releases of noble gases during and following an accident.

DESIGN BASIS MAXIMUM RANGE

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter of air at standard temperature and pressure (STP) for monitors employing beta radiation detector (Note: 1R/hr @1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gases changes) should be taken into account.

10^5 μCi/cc - Undiluted containment exhaust gases (e.g., PWR reactor building purge, PWR drywell purge through the standby gas treatment system).

- Undiluted PWR condenser air removal system exhaust.

10^4 μCi/cc - Diluted containment exhaust gases (e.g., > 10:1 dilution, as with auxiliary building exhaust air).

- BWR reactor building (secondary containment) exhaust air.

- PWR secondary containment exhaust air.

TABLE II.F.1-1

(CONTINUED)

10^3 μCi/cc	-	Buildings with systems containing primary coolant or primary coolant offgases (e.g., PWR auxiliary buildings, BWR turbine buildings).
	-	PWR steam safety valve discharge, atmospheric steam dump valve discharge.
10^2 μCi/cc	-	Other release points (e.g., radwaste buildings, fuel handling/storage buildings).
REDUNDANCY	-	Not required; monitoring the final release point of several discharge inputs is acceptable.
SPECIFICATIONS	-	(None) Sampling design criteria per ANSI N13.1.
POWER SUPPLY	-	Vital instrument bus or dependable backup power supply to normal ac.
CALIBRATION	-	Calibrate monitors using gamma detectors to Xe-133 23 equivalent (1 R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.
DISPLAY	-	Continuous and recording as equivalent Xe-133 concentrations or μCi/cc of actual noble gases.
QUALIFICATION	-	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS	-	Offline monitoring is acceptable for all ranges of noble gas concentrations. Inline (induct) sensors are acceptable for 10^2 μCi/cc to 10^5 μCi/cc noble gases. For less than 10^2 μCi/cc, offline monitoring is recommended. Upsteam filtration (prefiltering to remove radioactive iodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident. For external mounted monitors (e.g., PWR main steam line), the thickness of the pipe should be taken into account in accounting for low-energy gamma radiation.

(3) location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

(4) assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and,

(5) the source of power to be used.

b. Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

TABLE II.F.1-1

HIGH-RANGE NOBLE GAS EFFLUENT MONITORS

REQUIREMENT - Capability to detect and measure concentrations of noble gas fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored.

PURPOSE - To provide the plant operator and emergency planning agencies with information on plant releases of noble gases during and following an accident. 23

DESIGN BASIS MAXIMUM RANGE

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter of air at standard temperature and pressure (STP) for monitors employing beta radiation detector (Note: 1R/hr @1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gases changes) should be taken into account.

10^5 μCi/cc - Undiluted containment exhaust gases (e.g., PWR reactor building purge, PWR drywell purge through the standby gas treatment system).

- Undiluted PWR condenser air removal system exhaust.

10^4 μCi/cc - Diluted containment exhaust gases (e.g., > 10:1 dilution, as with auxiliary building exhaust air).

- BWR reactor building (secondary containment) exhaust air.

- PWR secondary containment exhaust air.

TABLE II.F.1-1

(CONTINUED)

10^3 μCi/cc	-	Buildings with systems containing primary coolant or primary coolant offgases (e.g., PWR auxiliary buildings, BWR turbine buildings).
	-	PWR steam safety valve discharge, atmospheric steam dump valve discharge.
10^2 μCi/cc	-	Other release points (e.g., radwaste buildings, fuel handling/storage buildings).
REDUNDANCY	-	Not required; monitoring the final release point of several discharge inputs is acceptable.
SPECIFICATIONS	-	(None) Sampling design criteria per ANSI N13.1.
POWER SUPPLY	-	Vital instrument bus or dependable backup power supply to normal ac.
CALIBRATION	-	Calibrate monitors using gamma detectors to Xe-133 equivalent (1 R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.
DISPLAY	-	Continuous and recording as equivalent Xe-133 concentrations or μCi/cc of actual noble gases.
QUALIFICATION	-	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS	-	Offline monitoring is acceptable for all ranges of noble gas concentrations. Inline (induct) sensors are acceptable for 10^2 μCi/cc to 10^5 μCi/cc noble gases. For less than 10^2 μCi/cc, offline monitoring is recommended. Upsteam filtration (prefiltering to remove radioactive iodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident. For external mounted monitors (e.g., PWR main steam line), the thickness of the pipe should be taken into account in accounting for low-energy gamma radiation.

23

ATTACHMENT 2 SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

Clarification

1. Licensees shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates to meet the requirements of the enclosed Table II.F.1-2. Licensees shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition. If gross gamma radiation measurement techniques are used, then provisions shall be made to minimize noble gas interference. 23

2. The shielding design basis is given in Table II.F.1-2. The sampling system design shall be such that plant personnel could remove samples, replace sampling media and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of GDC 19 of 5 rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

3. The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of ± 20%. Further departure from the isokinetic condition need not be considered in design. Corrections for non-isokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969 may be considered on an ad hoc basis.

4. Effluent streams which may contain air with entrained water, e.g., air ejector discharge, shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

TABLE II.F.1-2

SAMPLING AND ANALYSIS OR MEASUREMENT OF HIGH-RANGE RADIOIODINE AND PARTICULATE EFFLUENTS IN GASEOUS EFFLUENT STREAMS

EQUIPMENT	-	Capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The capability to sample and analyze for radioiodine and particulate effluents is not required for PWR secondary main steam safety valve and dump valve discharge lines.
PURPOSE	-	To determine quantitative release of radioiodines and particulates for dose calculation and assessment.
DESIGN BASIS SHIELDING ENVELOPE	-	10^2 μCi/cc of gaseous radioiodine and particulates, deposited on sampling media; 30 minutes sampling time, average gamma energy (E) of 0.5 MeV.
SAMPLING MEDIA	-	Iodine > 90% effective adsorption for all forms of gaseous iodine.
23	-	Particulates > 90% effective retention for 0.3 micron (μ) diameter particles.
SAMPLING CONSIDERATIONS	-	Representative sampling per ANSI N13.1-1969.
	-	Entrained moisture in effluent stream should not degrade adsorber.
	-	Continuous collection required whenever exhaust flow occurs.
	-	Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.
ANALYSIS	-	Design of analytical facilities and preparation of analytical procedures shall consider the design basis sample.
	-	Highly radioactive samples may not be compatible with generally accepted analytical procedures; in such cases, measurement of emissive gamma radiations and the use of shielding and distance factors should be considered in design.

ATTACHMENT 3 CONTAINMENT HIGH-RANGE RADIATION MONITOR

Position

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Clarification

1. Provide two radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1-3.

2. The specification of 10^8 rad/hr in the above position was based on a calculation of post-accident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments but gamma-sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 10^7 R/hr. 23

3. The monitors shall be located in containment(s) in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

4. For BWR Mark III containments, two such monitoring systems should be inside both the primary containment (drywell) and the secondary containment.

5. The monitors are required to respond to gamma photons with energies as low as 60 keV and to provide an essentially flat response for gamma energies between 100 keV and 3 MeV, as specified in Table II.F.1-3. Monitors that use thick shielding to increase the upper range will underestimate post-accident radiation levels in containment by several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

TABLE II.F.1-3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

REQUIREMENT - The capability to detect and measure the radiation level within the reactor containment during and following an accident.

RANGE - 1 rad/hr to 10^8 rads/hr (beta and gamma) or alternatively 1 R/hr to 10^7 R/hr (gamma only).

RESPONSE - 60 keV to 3 MeV photons, with linear energy response ± 20%) for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information.

REDUNDANT - A minimum of two physically separated monitors (i.e., monitoring widely separated spaces within containment).

DESIGN AND QUALIFICATION - Category 1 instruments as described in Appendix B, except as listed below.

SPECIAL CALIBRATION - In situ calibration by electronic signal substitution
23 is acceptable for all range decades above 10 R/hr. In situ calibration for at least one decade below 10 R/hr shall be by means of calibrated radiation source. The original laboratory calibration is not an acceptable position due to the possible differences after in situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.

SPECIAL ENVIRONMENTAL QUALIFICATIONS - Calibrate and type-test representative specimens of detectors at sufficient points to demonstrate linearity through all scales up to 10^6 R/hr. Prior to initial use, certify calibration of each detector for at least one point per decade of range between 1 R/hr and 10^3 R/hr.

ATTACHMENT 4 CONTAINMENT PRESSURE MONITOR

Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 lb/in.^{2}g for all containments.

Clarification

1. Design and qualification criteria are outlined in Appendix B.

2. Measurement and indication capability shall extend to 5 $lb/in.^2a$ for subatmospheric containments.

3. Two or more instruments may be used to meet requirements. However, instruments that need to be switched from one scale to another scale to meet the range requirements are not acceptable.

4. Continuous display and recording of the containment pressure over the specified range in the control room is required.

5. The accuracy and response time specifications of the pressure monitor shall be provided and justified to be adequate for their intended function.

ATTACHMENT 5 CONTAINMENT WATER LEVEL MONITOR

Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool. 23

Clarification

1. The containment wide-range water level indication channels shall meet the design and qualification criteria as outlined in Appendix A. The narrow-range channel shall meet the requirements of Regulatory Guide 1.89.

2. The measurement capability of 600,000 gallons is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

3. Narrow-range water level monitors are required for all sizes of sumps but are not required in those plants that do not contain sumps inside the containment.

4. For BWR pressure-suppression containments, the emergency core cooling system (ECCS) suction line inlets may be used as a starting reference point for the narrow-range and wide-range water level monitors, instead of the bottom of the suppression pool.

5. The accuracy requirements of the water level monitors shall be provided and justified to be adequate for their intended function.

ATTACHMENT 6 CONTAINMENT HYDROGEN MONITOR

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

Clarification

1. Design and qualification criteria are outlined in Appendix B.

2. The continuous indication of hydrogen concentration is not required
23 during normal operation.

   If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of safety injection.

3. The accuracy and placement of the hydrogen monitors shall be provided and justified to be adequate for their intended function.

RESPONSE

The implementation of the NUREG-0660, NUREG-0694, and NUREG 0737 Action Plan Item II.F.1 has resulted in six separate action items:

1. Radiological Noble Gas Effluent Monitors

2. Radioiodine and Particulate Effluents Monitors

3. Containment Radiation Monitors

4. Containment Pressure Indication

5. Containment Water Level Indication

6. Containment Hydrogen Indication

The six action items are addressed in the following sections and their implementation fulfills the NRC requirement for additional accident monitoring instrumentation.

A human factors analysis of the control room is being conducted as part of the control room design review discussed in item I.D.1.

Technical Specification 3/4.3.3.6 on accident monitoring instrumentation will include the high range containment radiation monitor, the containment pressure monitors, the containment water level monitors and the containment hydrogen monitors.

1.0 RADIOLOGICAL NOBLE GAS EFFLUENT MONITORS

1.1 GENERAL DESCRIPTION

FSAR section 11.5 provides detailed descriptions of the effluent monitors installed in San Onofre Units 2&3. The following additional monitors have been added to specifically address the NRC requirements for radiological noble gas effluent monitors (NUREG-0578, Section 2.1.8.b, NUREG-0660, 23
Section II.F.1, NUREG-0694, and NUREG-0737, Section II.F.1).

1.1.1 WIDE-RANGE EFFLUENT MONITOR

All potential gaseous accident release paths are monitored by the wide-range effluent monitors. The set of wide-range effluent gas monitors to be provided by SCE, per unit, consists of one monitor mounted on the steam jet air ejector exhaust stack (location shown in figure II.F.1-1), and one monitor with the capability of switching between containment purge and plant vent stack, mounted on the 63'6" level of the penetration area (figure II.E.1-2). Each wide-range effluent monitor is designed for a range of 10^{-7} μCi/cc to 10^{5} μCi/cc.

1.1.2 MAIN STEAM LINE MONITOR

Two area radiation monitors with overlapping ranges are mounted in close proximity to each steam line. These monitors measure direct dose rate from the main steam line to quantify effluent from the atmospheric dump, main steam relief valves, and auxiliary feedwater pump discharge.

1.2 DESIGN CRITERIA

23 The noble gas effluent monitors are designed to meet the recommendations of NUREG-0578, NUREG-0660, NUREG-0694, and NUREG-0737. FSAR section 11.5 provides the design basis for effluent monitors. Design criteria relating to minimizing radiation exposure is discussed in section II.B.2.

The additional monitor design criteria are as follows:

1.2.1 WIDE-RANGE EFFLUENT MONITOR

A. Each monitor is designed for a radiation range of 10^{-7} µCi/cc to 10^{5} µCi/cc (Xe-133). The range capacity of individual monitors overlap by at least a factor of 10 (refer to figure II.F.1-3).

B. Monitors provide digital readout and continuous recording of effluent in the control room.

C. One annunicator, per monitor, is provided on the main control board to alert personnel of high radiation levels.

D. All filter grab sample and flow switching controls are located in the control room.

E. Filters are designed with materials which minimize the adsorption of noble gases.

F. The sample conditioning skid is equipped with grab sample capability. All other filters are removable, with their lead shields, from the skid for analysis.

G. Heat tracing, moisture separator and cooling loops are used as needed to reduce the relative humidity and temperature of the sample. This ensures sample temperature and humidity will not degrade sample filter or detector.

H. Monitors are located in areas so that the peak background following a postulated accident will not affect the detector.

23 In addition, the following design criteria are applicable to the wide-range effluent gas monitors:

A. Quality Class II

B. Seismic Category I

C. Electrical Class 1E

D. Isokinetic probes are installed in accordance with ANSI N13.1-1969. The probes are Quality Class II and Seismic Category I.

E. Sample piping and tubing is mounted according to Quality Class III, Seismic Category I, and Project Class W for the containment stacks, Quality Class III, Seismic Category II and Project Class J for the steam jet air ejector stack. 23

F. The valves employed in the isokinetic flow switching manifold, for the containment purge and continuous exhaust vent stack monitor, are Quality Class II, Seismic Category I, and Project Class P.

G. A weather-proof enclosure around the sampling conditioning and detector skid, for the steam jet air ejector monitor, is Seismic Class II. 23

1.2.2 MAIN STEAM LINE MONITOR

A. Continuous readout and recording of all main steam line monitors is provided in the control room.

B. Quality Class 2

C. Seismic Category I

D. Electrical Class 1E

1.3 DETAILED DESCRIPTION

1.3.1 WIDE-RANGE EFFLUENT MONITOR

Due to the high gamma dose rates occurring as a result of the radioactive concentrations during a high level release, it is necessary to separate the monitor into several assemblies. As a result, there are five major components to a monitor: 1. Isokinetic Samplers and Flow Transmitter, 2. Sampler Conditioning, 3. Sample Detection, 4. Electronics, and 5. Readouts. These components are described individually below. Mechanical components 1 through 3 are an open frame designed to allow cooling by natural convection. All plumbing and piping is stainless steel. All connections are leak tested.

1.3.1.1 Isokinetic Sampler and Flow Transmitter

There are two sets of isokinetic samplers provided for each stack. Each isokinetic consists of two sets of isokinetic nozzles, one for sampling during low concentration levels and one for sampling during high concentration levels. Both sets of nozzles are mounted on a common sample gantry in the duct. The low concentration level nozzle operates at approximately 2 ft^3/min., whereas the high concentration level nozzles operate at approximately 0.06 ft^3/min to minimize activity buildup.

Each monitor has a flow transmitter for each of the vents being sampled. The flow transmitter is used to measure the total vent flow rate. This signal is then used to control the sample flow rate so as to maintain true isokinetic sampling conditions. Additionally, the vent flow rate signal allows the operator to display the effluent radiation level in units of mCi/sec (Xe-133).

1.3.1.2 <u>Sample Conditioning</u>

This assembly is connected to the isokinetic samplers and provides for both collection of particulate and iodine samples and sample conditioning. There are two sets of sample collectors - one for low range operation and one for high range operation. Filter holders are connected via quick disconnect fittings to allow rapid retrieval of particulate and iodine samples for isotopic analysis. Sample conditioning is provided to prevent contamination of the sample detection assembly by filtering out a major proportion of the iodines and particulates. Without this sample conditioning, the detectors would be contaminated and would remain upscale even when actual radioactive gas levels decrease. To provide enough filtering material to contain the iodines and particulates for the duration of the measurement period, special multiple filters are employed. Since, in a high range sample condition, high activity will be accumulated on these filters, each set of filters is surrounded with a two inch 4 lead shield to reduce exposure dose rates. The design basis for the filter shielding is 10^{5} μCi/cc of noble gas and 10^{2} μCi/cc iodines leaving the stack at the design flowrate. The filter is modeled as a flow thru system assuming a 99% particulate filter efficiency and 95% silver xeolite efficiency. The model considers buildup of daughter products, as well as the actual gamma
23 energies for each isotope. Figure II.F.1-9 shows the buildup of activity on the shielded filters. Figure II.F.1-10 shows the dose rate versus time for these filters.

This design basis exceeds the shielding basis presented in Table II.F.1-2 of NUREG-0737.

The entire sample conditioning assembly is designed to require minimum maintenance. The dimensional envelope of the assembly is approximately 30 in. H. x 48 in. W. x 30 in. D. It weighs approximately 1200 lbs.

1.3.1.3 <u>Sample Detection</u>

This assembly contains three radioactive gas detectors that monitor the sample discharged from the sample conditioning assembly. Their range of detection is 10^{-7} to 10^{5} mCi/cc (Xe-133).

This assembly also contains the necessary pumps, flow control valves, flowmeters, etc. Each detector has a solenoid-actuated checksource to verify proper operation. The 12 decades of noble gas concentrations are monitored by the three detectors with at least one decade overlap between ranges of the individual detectors. The low-range detector utilizes a plastic scintillator, whereas the mid- and high-range detectors are solid-state and are specially selected for this application.

There are two flow paths through the detectors. During normal operation, only the low-range detector is used and the mid-range and high-range detectors are bypassed. As the low-range detector begins to saturate, the flow path is automatically changed to the mid- and high-range detectors and the low-range detector is purged. This prevents contamination of the low-range detector so that it will be available when it is automatically returned to service to measure radioactive gas concentrations after they have returned to low levels.

The assembly is designed for ease of servicing. The dimensional envelope of the assembly is approximately 57 in. H x 48 in. W x 30 in. D and it weighs approximately 5,300 lbs. All electrical connections are provided to terminal boards and coax connectors inside a junction box mounted on the single skid assembly.

The detectors used in the wide range effluent monitor will have a primary laboratory calibration report. This calibration report establishes the linearity of the detector. Field calibration of these detectors involves using a single calibration source which when placed in the proper geometry relative to the detector verifies a single point on the detector calibration curve. The frequency of field calibration for the wide range effluent monitor is every refueling. 23

1.3.1.4 Electronics

The monitor is completely operated by the Model RM-80 Microprocessor. The RM-60 performs flow control, valve actuations, engineering conversions, and other calculations and control functions. It is located in the control room and contains the microprocessor memory, high-voltage power supplies, preamplifiers, battery back-up power, etc. Mounted under the RM-80 is a junction box for termination of cables between the RM-80 and the rest of the monitor.

1.3.1.5 Readouts

The readouts are designed for control room panel mounting and consist of a digital module (Model RM-23) and a chart recorder. The RM-23 is microprocessor-based and provides a display of all monitor parameters. Monitor parameters displayed include channel activity flow rates, alarm status, check-source actuation, valving control, etc. The RM-80 also maintains history files of twenty-four 10-min, twenty-four 1-hr, and twenty-eight 1-day averages of channel activity that are available for recall via the RM-23. Power for each RM-23 will be provided by its associated RM-80.

1.3.1.6 System Operation

During normal operation both the Unit 2 and Unit 3 monitors will be aligned to sample the plant vent stack as shown on figure II.F.1-11. With this alignment, one of the filters (c) for both the low and high range flow paths will be assigned to the plant vent stack.

For a postulated accident in Unit 2, the Unit 2 containment purge system will be isolated as described in FSAR subsection 6.2.4. The operator, using radiological information provided by the high range incontainment monitors (FSAR subsection 12.3.4), containment airborne monitors (FSAR paragraph 11.5.2.1.4.5), emergency radiation monitoring system (FSAR subsection 12.3.4) and postaccident sample system (FSAR subsection 9.3.6) would decide when to initiate containment purge. Prior to purging the containment, the wide range effluent monitors would be aligned as shown on figure II.F.1-12.

With the post-accident configuration as shown on figure II.F.1-12, the Unit 3 wide range effluent monitor would be sampling from the plant stack. Due to the mixing of the building ventilation paths, this Unit 3 monitor will measure approximately one-half of the total plant stack release. The ventilation mixing is discussed in the response to FSAR question 321.12.

23 The post-accident configuration will result in the following activity being on the filters for a postulated accident in Unit 2:

1. Unit 2 wide range effluent monitor C filter will contain one-half of the total particulate and iodine from plant stack releases during normal operations up to time of purge of Unit 2 containment following a postulated accident.

2. Unit 2 wide range effluent monitor B filter will contain particulate and iodine from containment purge releases following a postulated Unit 2 accident.

3. Unit 3 wide range effluent monitor C filter will contain approximately one-half of the total particulate and iodine from plant stack releases during normal operations up to time of purge of Unit 2 containment following a postulated accident.

4. Unit 3 wide range effluent monitor B filter will contain approximately one-half of the total particulate and iodine from plant stack releases during purging of Unit 2 containment following postulated Unit 2 accident.

5. At the operator's discretion, a grab sample (A) can be taken. The controls and recorders for the wide range effluent monitors are located in the control room panels 2L-405 and 3L-405 as described in FSAR paragraphs 11.5.2.1.1.5 and 11.5.2.1.1.6.

At the time of switchover to the monitor configuration for containment purge the continuous recorders in the control room will be marked to indicate switchover has occurred. 23

1.3.2 MAIN STEAM LINE MONITOR

The main steam line monitors are area radiation monitors located adjacent to the main steam line. Each steam line is monitored by two area radiation monitors with overlapping ranges of 0.1 to 10^4 mR/h and 0.1 to 10^4 R/h.

Calculational methods will be employed to quantify radiological releases based on monitor dose rates. In order to obtain concentrations of 10^{-1} to 10^3 μCi/cc of Xe-133 in the main steam line a large primary to secondary leak must be present coincident with a large amount of fuel failure. Present with Xe-133 will be other nuclides. Based on the ratios of isotopic composition in the fuel, and steam generator reduction 23
factors of 1 for noble gases, 0.01 for iodines and infinite for other isotopes relative main steam line concentrations can be determined. A QAD computer model considering detector geometry and detector vs energy response is used to determine dose rates from steam line concentrations. A detailed description of this methodology will be described in response to FSAR question 321.14. Figures II.F.1-4 through II.F.1-6 show the locations of these monitors.

The main steam line detectors are shielded by 4-inches of steel. This steel acts as a collimator so the detector looks at a known main steam line geometry. The steel reduces the potential background from the atmospheric dump and relief valve piping. The detector is also located next to a 2 foot concrete supporting wall. This means that the detector is shielded from the containment atmosphere by 6-feet of concrete and 4-inches of steel.

The geometry of the main steam line detector and potential radiation streaming through the steam line containment penetration makes elimination of all background from containment atmosphere impossible. Correction for containment background is based on the symmetry of the main steam line detectors relative to containment. This symmetry is illustrated in Figure II.F.1-4. Because of this symmetry, the background radia- 23
tion from the containment atmosphere will be identical for both sets of detectors. Thus, one set of detectors serves as a background measurement for the other.

The main steam line monitor instrument readings will be corrected for background by comparing the two sets of monitor readings and subtracting the appropriate monitor value.

Calibration of the main steam line detectors involves the use of an RT-10 calibrator. The RT-10 calibrator contains 10m Ci of Cs-137. The calibrator is hung on the detector bracket. The geometry of the calibrator and source can be changed so as to obtain two different radiation levels within the range of each detector. The frequency of calibration for the main steam line detectors is every refueling.

1.4 MEETING NRC REQUIREMENT

The monitors discussed in FSAR section 11.5 in conjunction with the new wide-range effluent and main steam line monitors fulfill the NRC requirements as outlined in NUREG-0578, Clarification to NUREG-0578 (NRC letter
23 Nov. 9, 1979), NUREG-0660, NUREG-0694 and NUREG-0737. These monitors will be installed prior to January 1, 1982.

1.5 REFERENCE

Revised FSAR section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems.

2.0 RADIOIODINE AND PARTICULATE EFFLUENT MONITORS

2.1 GENERAL DESCRIPTION

FSAR section 11.5 provides detailed descriptions of the radioiodine and particulate effluent monitors installed in San Onofre Units 2&3. The wide-range effluent monitor has been added to specifically address the NRC requirements for radioiodine and particulate effluents (NUREG-0578 section 2.1.8.b, NRC letters of Sept. 27 and Nov. 9, 1979, NUREG-0660,
23 NUREG-0694, and NUREG-0737).

The wide-range effluent monitor is provided with grab sample cartridges for collection of particulate and iodine samples. This sample will be analyzed onsite. The wide-range gas monitor sample conditioning filters are also removable. This supplements the grab sample collection capability.

The location of the wide-range effluent monitor is discussed in section 1.1.1.

2.2 DESIGN CRITERIA

The design criteria and design basis for the particulate and iodine samples is described in section 1.2.1.

2.3 DETAILED DESCRIPTION

Section 1.3.1 provides a detailed description of the wide-range effluent monitor and the particulate and iodine sampling capability.

2.4 MEETING NRC REQUIREMENT

The monitors discussed in FSAR section 11.5, in conjunction with the new wide-range effluent monitor, satisfy the criteria for sampling radioiodine and particulate as described in NUREG-0737. These monitors will be 23
installed prior to January 1, 1982.

2.5 REFERENCE

Revised FSAR section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems.

3.0 CONTAINMENT RADIATION MONITORS

3.1 GENERAL DESCRIPTION

FSAR subsection 12.3.4 provides descriptions of the emergency radiation monitoring system (ERMS) installed in San Onofre Units 2 & 3. Additional high range in-containment monitors have been added to supplement the ERMS 23
monitors. These monitors are designed to satisfy the NRC requirements for containment radiation monitors (NUREG-0578 Section 2.1.8.b, NUREG-0660, NUREG-0694, and NUREG-0737 Section II.F.1).

The new high range radiation monitors consist of two (channels A and B) gamma-sensitive radiation detectors for each containment. Each monitor has a range of 10^0 to 10^8 rads/h. One of the detectors is mounted at elevation 93', adjacent to the polar crane access ladder. The other detector is mounted on top of the secondary shield wall at elevation 98'. Figure II.F.1-7 shows these locations.

The high range radiation detectors are suitable for use in a containment environment. Power and readout modules are located in the control room.

3.2 DESIGN CRITERIA

The ERMS in conjunction with the high range in-containment radiation monitors are designed to meet the specific design criteria of Table II.F.1-3 of NUREG-0737. The design criteria for the ERMS monitors 23
is provided in FSAR subsection 12.3.4.

Specific design criteria for the high range in-containment radiation monitors are:

A. Radiation range 10^0 to 10^8 rads/h.

B. Two physically separated channelized detectors per containment will be installed.

C. Continuous readout of both channels and a continuous recording of one channel will be provided in the control room.

D. Detectors are powered through channels A&B which are redundant and are designed so that failure in one channel will not affect the other channel.

E. Detectors are mounted so that containment equipment and structures will not shield the detectors from the source.

F. One annunciator per unit on the main control board to alarm when either channel has a high, high-high radiation level and/or channel failure.

G. Components outside of containment are designed for an ambient temperature range of 40F to 120F. In-containment components are qualified for LOCA. (Design qualification temperature 350F. This is in excess of the 300F requirement listed in FSAR table 3.11-1.)

H. Components outside containment are designed for atmospheric pressure. Components inside the containment are designed to withstand 70 lb/in^2 gage. This exceeds the 60 lb/in^2 requirement of FSAR table 3.11-1.

I. Detectors (within containment) can withstand humidity of 100% (saturated steam). The electronics (control room) were designed for up to 95% humidity. This meets the criteria of FSAR table 3.11-1.

J. Containment components are designed to withstand a chemical spray of 2350 ppm Boron and a NaOH pH value of 9-10. This meets the criteria of FSAR table 3.11-1.

23 K. Detector energy dependence ±20% from 80 KeV to 3 MeV.

In addition, the following design criteria are applicable to the high range radiation monitor:

- Quality Class II
- Seismic Category I
- Electrical Class 1E

- Single Failure Design
- Operable From Emergency Power
- Can Withstand Normal And Post-LOCA Environmental Conditions Stated In FSAR Section 3.11.

3.3 DETAILED DESCRIPTION

FSAR subsection 12.3.4 provides a detailed description of the ERMS monitors. Each high range detector has two adjustable alarm setpoints that provide input into one out of two logic circuits. The logic circuits produce a high alarm, high-high alarm, and/or a channel failure alarm, as appropriate, on the main panel annunciator. The channel-failure alarm is actuated upon loss of power, high voltage, or signal from the detector. Automatic self-testing is provided to continuously verify detector operation.

The safety function of the high range radiation monitor is to provide post-accident monitoring capability sensitive to 60 keV photons and to a radiation level of up to 10^8 rad/h. The two channels, A and B, are redundant and are designed so that failure in one channel does not affect the performance of the other channel. The monitor is designed and qualified in accordance with the applicable portion of IEEE 323-1974. The detector is a General Atomic Model RD-23 (Reuter Stokes RS- C3-1006-201) and associated RP-2C readout module.

The initial calibration for the high range in-containment monitor involves the use of a current source to generate a signal output. This is then verified by using a calibrated Cs-137 radiation source and standard geometry. 23
For field calibration an RT-11 calibrator is being developed. The RT-11 will contain a line source of approximately 100 millicuries/cc of Cs-137 providing a source of 1.2 to 1.5 R/h. This calibrator will be capable of being hung on the detector and will be shielded to reduce background levels. The frequency of calibration of the high range in-containment monitors is every refueling.

3.4 MEETING NRC REQUIREMENTS

The ERMS and high range radiation monitors satisfy the NRC requirements, as outlined in NUREG-0578, Clarification to NUREG-0578 (NRC letter Nov. 9, 1979), NUREG-0660, NUREG-0694, and NUREG-0737 to monitor in-containment radiation levels to 10^8 rads/h. The qualification of the high range radiation monitors meets the NRC design criterion as described in Table II.F.1-3 of NUREG-0737. 23
These monitors will be installed prior to fuel load.

3.5 REFERENCES

Revised FSAR sections 11.5, 12.3, and table 3.2-1.

4.0 CONTAINMENT PRESSURE INDICATION

4.1 GENERAL DESCRIPTION

FSAR section 7.2 provides a description of the existing containment pressure instrumentation system. The following added wide-wide range containment pressure measurement system consists of redundant gauge pressure transmitters whose signals of containment pressure will be continuously displayed within the control room. The range of the added system will be from 0 to 200 lb/in.^{2}g, in excess of three times the containment design pressure. This system is in addition to the existing containment pressure measurement system which has a range of -4 to 84 lb/in.^{2}g. Effectively, the new range of measurement of containment pressure is from -4 to +200 lb/in.^{2}g in
23 3 ranges; -4 to 20, -4 to 84 and 0 to 200 lb/in^{2}g, respectively.

4.2 DESIGN CRITERIA

The added containment pressure indicators meet the following design criteria as stipulated in NUREG-0737.

A. The instruments will be physically separated Class 1E pressure transmitters.

B. The signal of all channels will be continuously displayed in the control room. Continuous recording is provided for one channel for the entire range of pressure measurements.

C. The transmitters are located outside of the containment structure. They will sense the containment pressure through sensing lines penetrating the containment structure.

D. The transmitters will be environmentally qualified to the condition of their location, i.e., penetration area. All other equipment, power supplies, current-to-voltage converters, indicators, and recorders will be located in the control room complex.

E. The containment wide range pressure indication is provided by an electronic force-balance instrument and an electronic indicator
23 in the control room. The transmitter used is the same model used for the normal range containment pressure measurement. The accuracy of the instrument loop (±2% span) is acceptable for its intended use as the possible error is limited to 4.0 lb/in.^{2}g.

The response time of the NE11GM transmitter will be on the order of 0.4 seconds to reach 90% of its final reading for a 25% step change. The settling time (time in seconds it takes for the transmitter to settle out within ±2% of its final value is approximately an additional 0.4 seconds. This response time is adequate for its intended service as a post-accident monitor. 23

The channel response time for the narrow range and intermediate range pressure instruments is 0.9 sec. The instruments provide input to the RPS and ESFAS as described in FSAR sections 7.2 and 7.3.

In addition, the following design criteria are applicable:

All equipment, including power supplies, indicators, recorders, and converters will be:

- Class 1E
- Quality Class II
- Seismic Category I

4.3 MEETING NRC REQUIREMENTS

The containment pressure measurement system meets all the requirements as outlined in Clarification to NUREG-0737 with the exception of the low range limit which is specified as minus 5 $lb/in.^2g$. The SCE-installed pressure indicators have a lower limit measuring capability of minus 4 $lb/in.^2g$. These monitors will be installed prior to fuel load. This lower limit of the pressure measurement range is considered to be adequate since the maximum negative internal containment pressure is calculated to be 3.4 lb/in^2g (see response to NRC Question 022.34). 23

4.4 REFERENCE

FSAR section 7.2, Reactor Protective System, and FSAR section 7.5, tables 7.5-1, and 7.5-2. 23

5.0 CONTAINMENT WATER LEVEL INDICATION

5.1 GENERAL DESCRIPTION

A continuous indication of the containment water level will be provided in the control room by sensing the water level at the following locations:

- Containment Normal Sump
- Containment Emergency Sump
- Containment Periphery

Sufficient overlap exists between the three ranges of measurement to provide a continuous display over the entire range which covers elevation 10'2" to 29' of the containment (see figure II.F.1-8).

5.2 DESIGN CRITERIA

A. Physically separated (redundant) Class 1E transmitters will be provided for each location.

B. All channel signals will be continuously displayed in the control room.

C. Digital readout and continuous recording will be provided for one channel from each of the three location transmitters.

D. All transmitters will be LOCA qualified to the appropriate pressure, temperature, humidity, and radiation levels.

E. All the equipment, including power supply, indicator recorders and convertors, will be Class 1E.

23 F. The containment water level indications are provided by a transmitter, the level sensing unit, and an electronic indicator. The level sensing unit employs a float which moves with liquid level along a tubular stem containing a voltage divider. The float magnetically closes reed switches producing voltage drops at each 1/2 inch of float travel. Consequently, any inaccuracy is limited to 1/2 inch plus meter and other circuit tolerances. Overall channel accuracy is ±2% which is consistent with other post accident monitors.

In addition, the following design criteria are applicable:

- Quality Class II
- Seismic Category I
- Electrical Class 1E

5.3 MEETING NRC REQUIREMENTS

The SONGS 2 & 3 design complies with the NRC requirements presented in Clarification to NUREG-0578 (NRC letter Nov. 9, 1979). The three level sensors cover the range from the bottom to the top of the containment sump and in excess of an elevation equivalent to a 600,000 gallon capacity. The monitors will be installed prior to fuel load.

5.4 REFERENCE

None.

6.0 CONTAINMENT HYDROGEN INDICATION

6.1 GENERAL DESCRIPTION

FSAR subsection 6.2.5 provides a detailed description of the containment hydrogen monitoring system. The containment post-LOCA hydrogen monitoring system consists of two independent analyzer channels for each containment. Each channel includes a hydrogen sensor and a pressure transducer mounted within the containment and an electronics assembly mounted within a cabinet located in the control building. 23

Hydrogen concentration is displayed both at the cabinet and in the control room. High hydrogen concentration and system malfunction alarms are provided in the control room and are indicated by lights at the cabinet. Both channels can be initiated from either the control room or the cabinet.

Each channel has its own calibration system consisting of two gas bottles (one nitrogen, one nitrogen with 4% hydrogen) and solenoid operated valves controlled from the cabinet.

6.2 DESIGN CRITERIA

The containment post-LOCA hydrogen monitor system is qualified as a safety-related system and meets the requirements of NUREG-0737, with the exception that neither channel is recorded. 23

The design bases for the system are as follows:

A. Hydrogen concentration range is 0-10%.

B. Two physically separated channelized hydrogen sensors and pressure transducers are installed 90° apart in each containment. 23

C. A continuous readout of both channels is provided in the control room when the system is actuated.

D. The sensors and electronics assemblies are powered from channels A and B which are redundant and are designed to preclude failure in one channel from affecting the other channel.

E. One annunciator window is provided in the control room to alarm when either channel has a high hydrogen concentration. A separate window is provided to indicate a malfunction in either channel -- either from loss of power or sensor depletion.

F. Components within the containment are being qualified for the SONGS 2&3 LOCA profile.

In addition, the following design criteria are applicable:

- Quality Class II
- Seismic Class I
- Electrical Class 1E.

6.3 DETAILED DESCRIPTION

The containment post-LOCA hydrogen monitoring system consists of two electrically and physically separated analyzer trains. Each train consists of components inside and outside of the containment. The components within the containment, for one train, are a hydrogen sensor and a pressure transducer. A millivolt signal is produced as a function of hydrogen pressure and transmitted out of the containment to an electronics subassembly. Since the sensor is a partial pressure measuring device, an absolute pressure transmitter is mounted with the sensor and provides a pressure compensating signal to the electronics subassembly located outside of the containment.

The containment post-LOCA hydrogen monitoring system hydrogen sensors are located inside the containment building at elevation 76' on the outside of the secondary shield. This location provides the hydrogen sensor with access to the upper regions of the containment building and allows optimum monitoring of the containment atmosphere while maintaining accessibility in the event of sensor malfunction.

23 The accuracy of the containment post-LOCA hydrogen monitoring system is acceptable for the system's intended purpose, as the system is used to indicate hydrogen concentration and to verify removal of hydrogen by the hydrogen recombiners. The system performs no automatic initiation of other systems. The hydrogen recombiners will be started manually before the hydrogen concentration reaches 2%. The hydrogen monitor will alarm in the control room when hydrogen concentration reaches 3%. Considering system accuracy, there remains sufficient margin for assessment of action before the hydrogen purge need be initiated and before the lower flammability limit of 4% is reached.

The electronics subassemblies (one for each train) are mounted in one cabinet located in the control building, elevation 50 ft. The cabinet provides separation between the two electronics systems. Maintenance and calibration can be performed at the cabinet.

Each train has a completely adjustable high hydrogen alarm setpoint and a malfunction alarm.

The calibration gas supply is not safety-related. It provides separate zero and span gas to each hydrogen sensor through an arrangement of tubing and solenoid valves operated from the electronics cabinet.

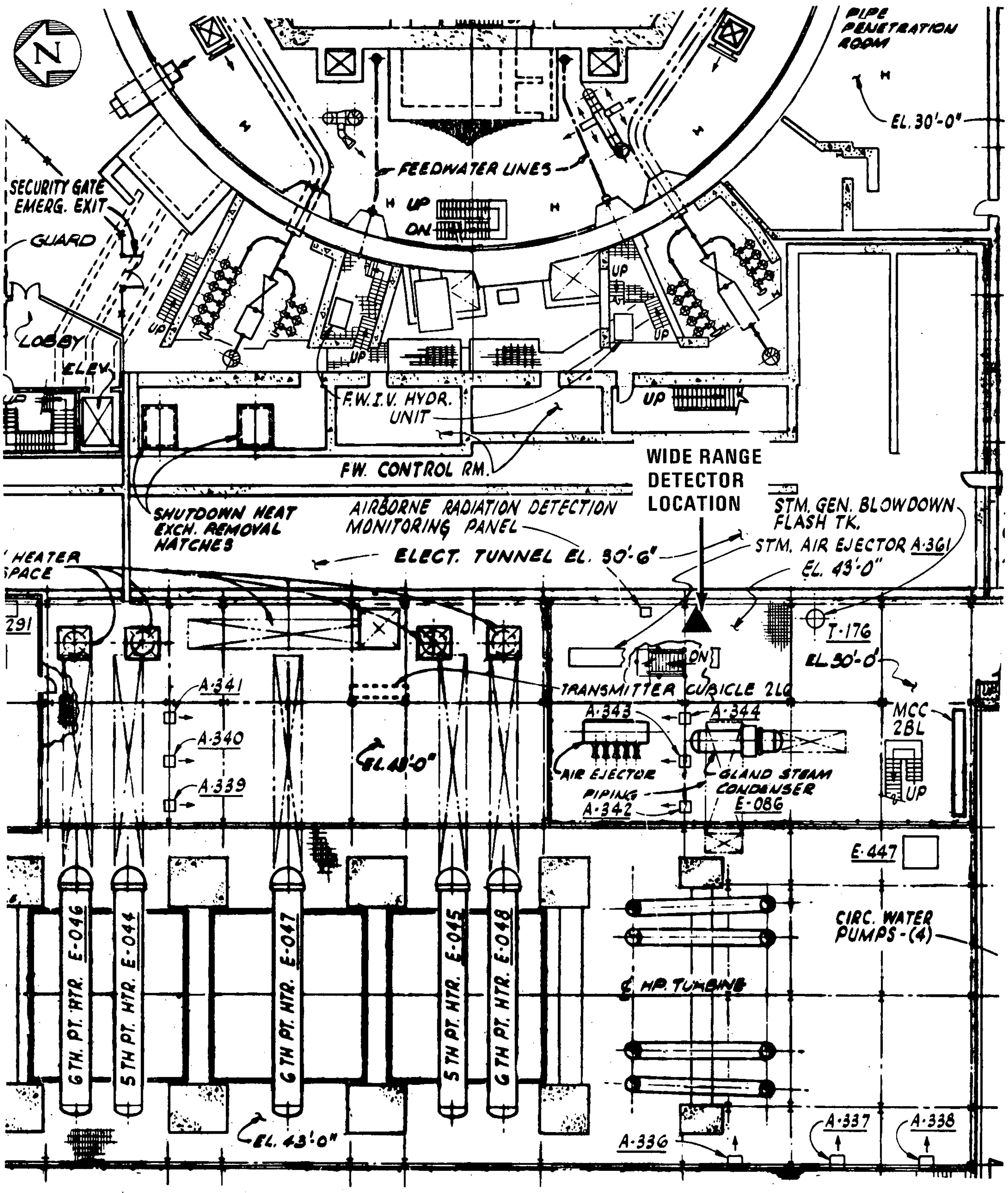
The safety function of the containment post-LOCA hydrogen monitoring system is to provide indication of the hydrogen concentration within the containment, post-LOCA.

6.4 MEETING NRC REQUIREMENTS

The containment post-LOCA hydrogen monitoring system satisfies the NRC requirement outlined in NUREG-0737, i.e. monitoring hydrogen concentration over the range of 0-10% under both positive and negative ambient pressure. 23

6.5 REFERENCE

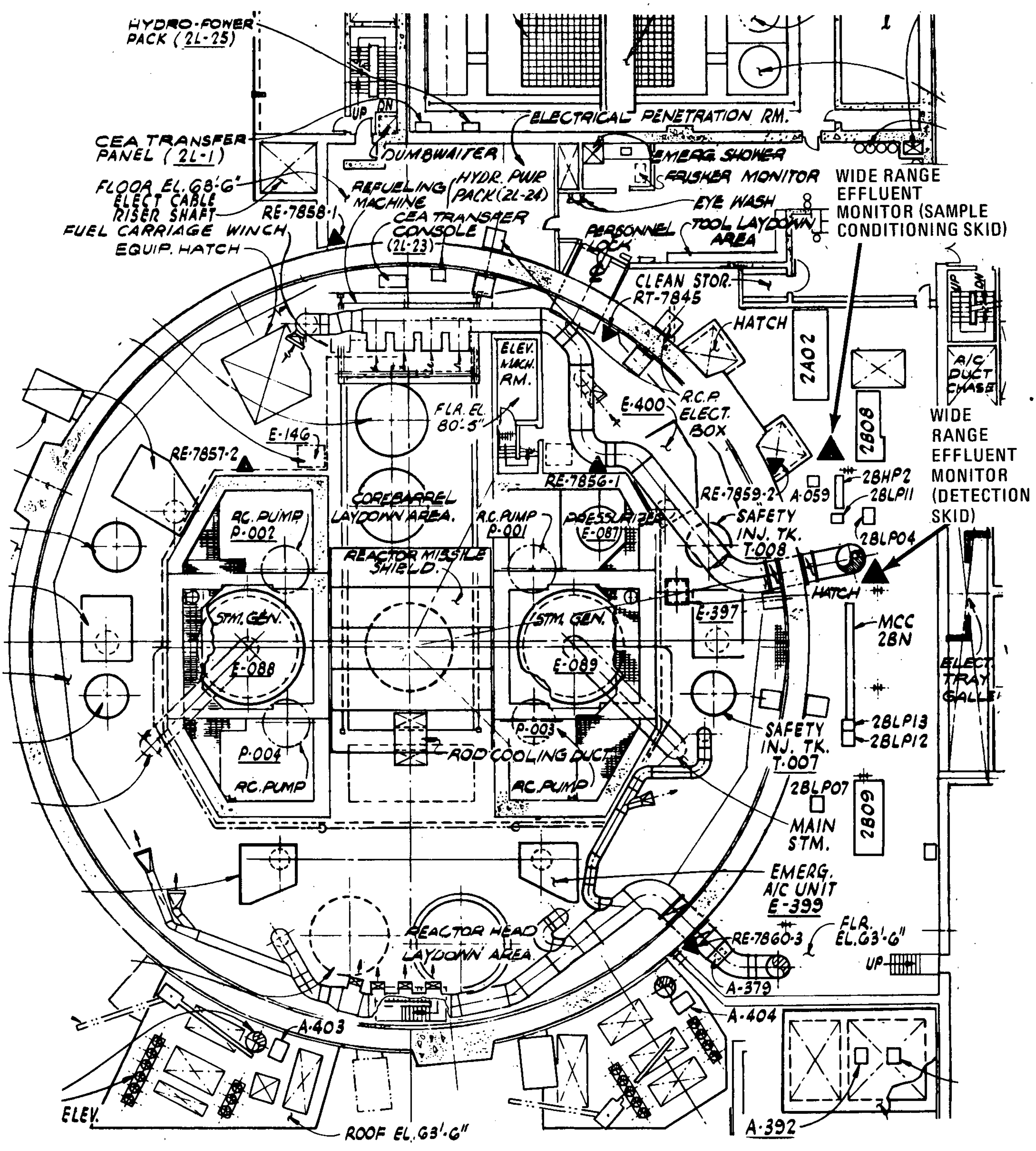
FSAR subsection 6.2.5, Combustible Gas Control in Containment, has been revised to be consistent with the above. 23



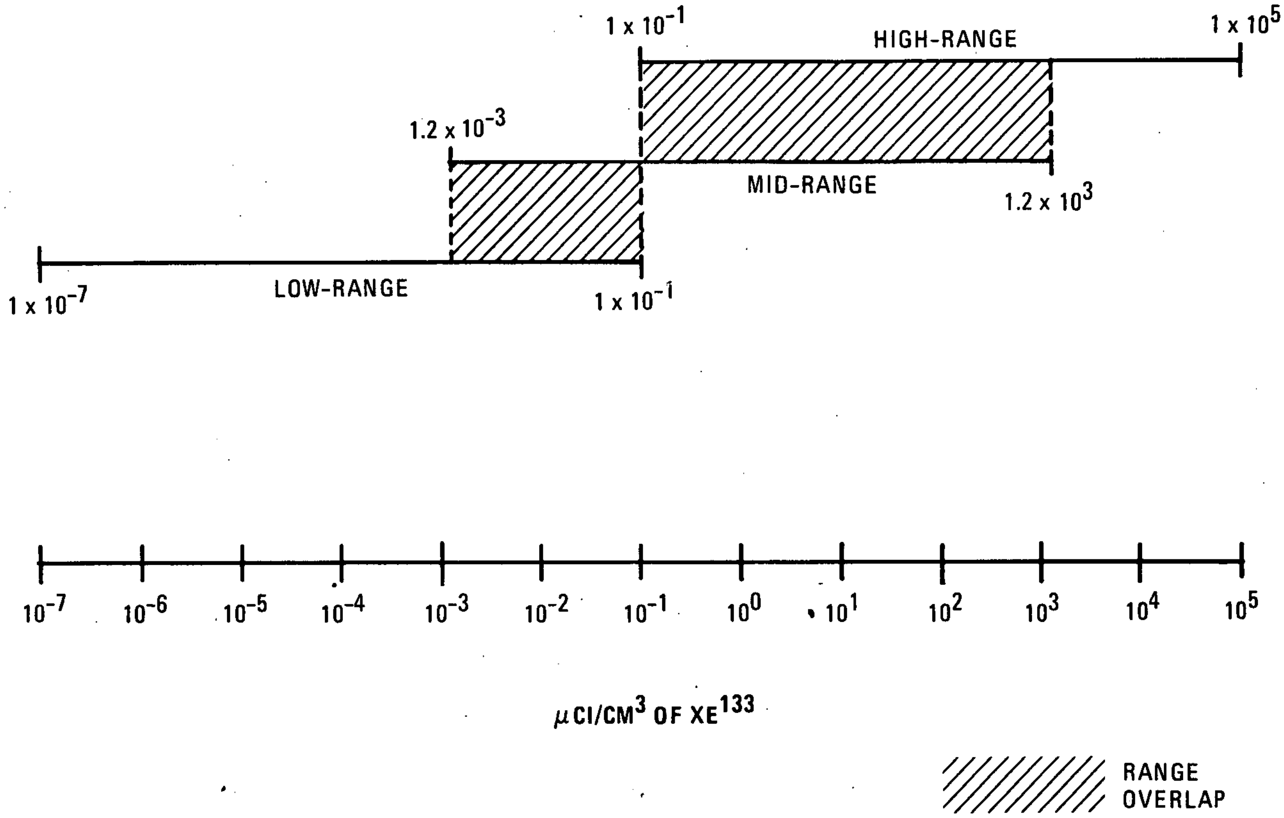
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

STEAM JET AIR EJECTOR WIDE RANGE EFFLUENT MONITOR LOCATION (30' EL. TURBINE BLDG.)(UNIT 2 SHOWN)

Figure II.F.1-1



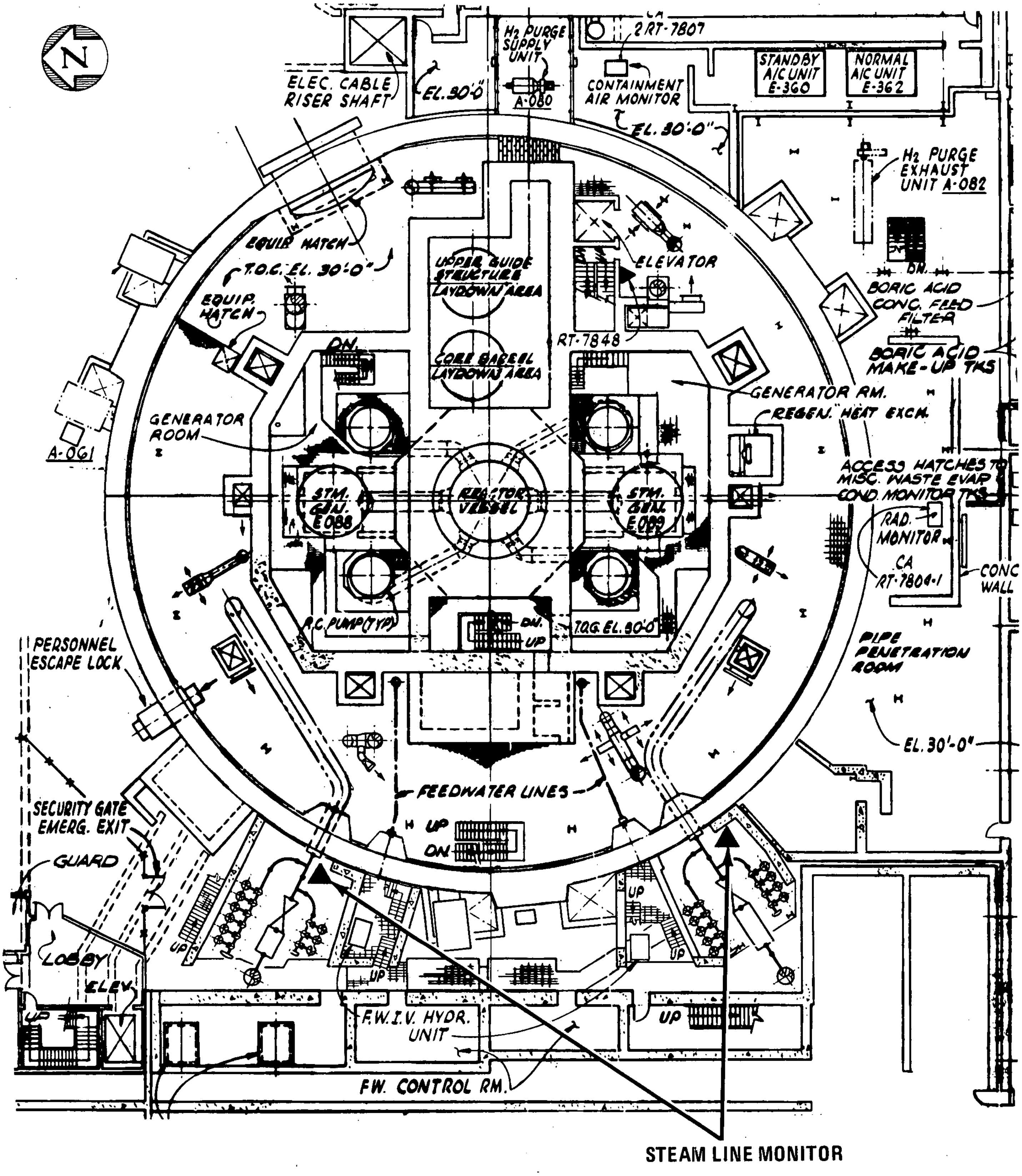
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
WIDE RANGE EFFLUENT MONITOR CONTAINMENT PURGE AND PLANT VENT STACK LOCATION (63'-6" LEVEL PENETRATION AREA) (UNIT 2 SHOWN)
Figure II.F.1-2



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

DETECTOR RANGES,
WIDE-RANGE GAS MONITOR

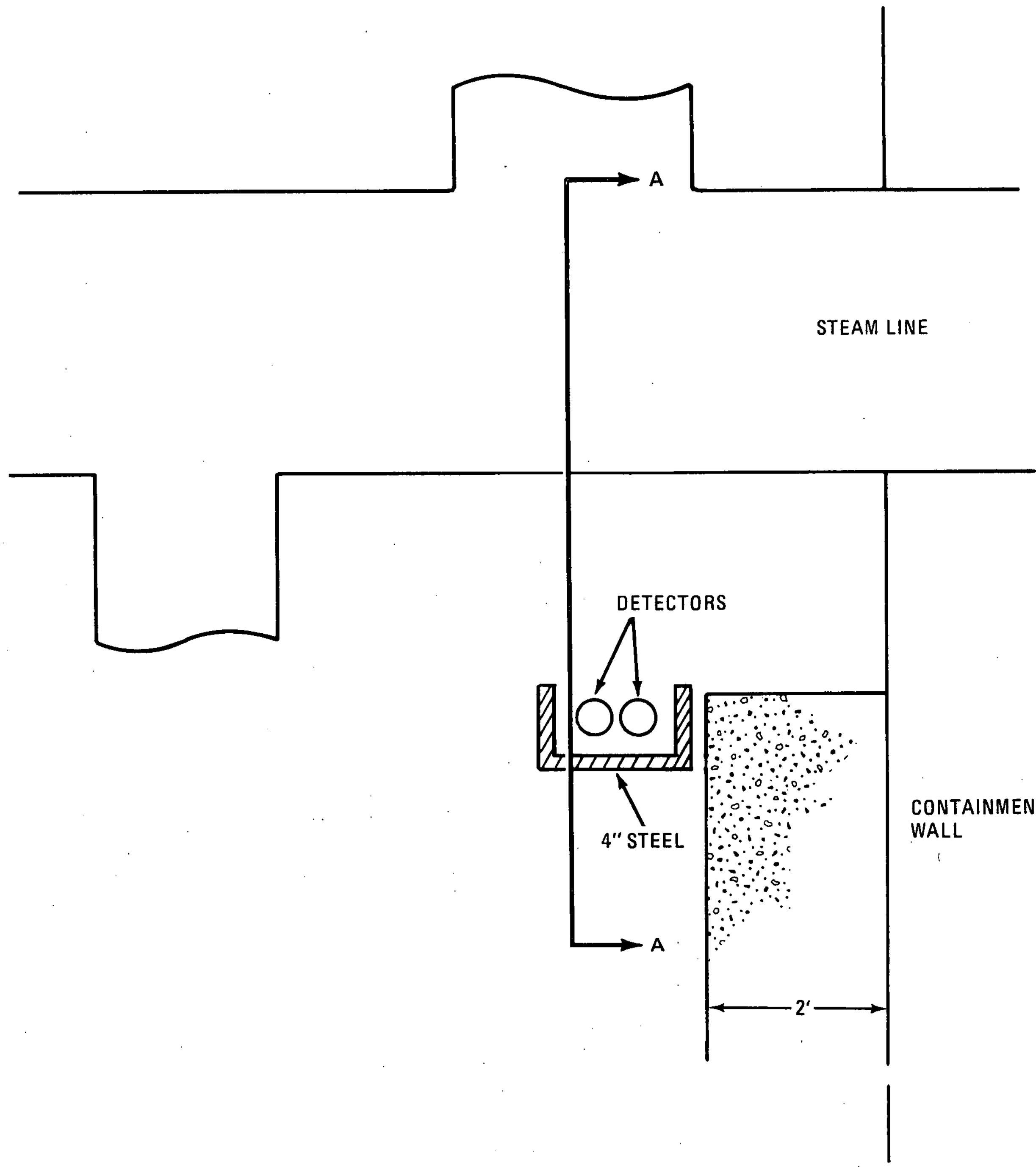
Figure II.F.1-3



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

MAIN STEAM LINE RADIATION MONITOR
(MSIV ROOM, UNIT 2 SHOWN)

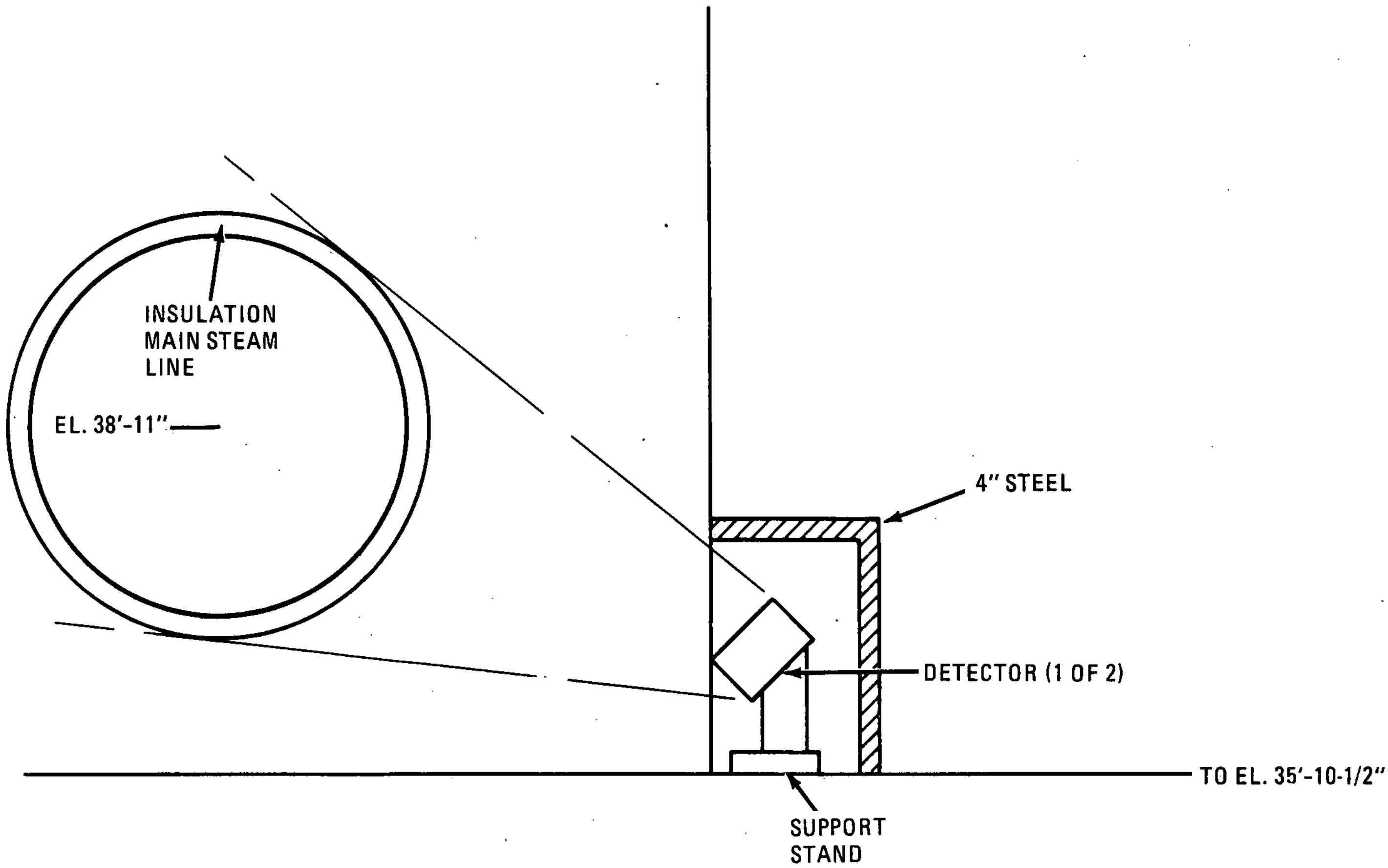
Figure II.F.1-4



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

MAIN STEAM LINE
MONITOR

Figure II.F.1-5

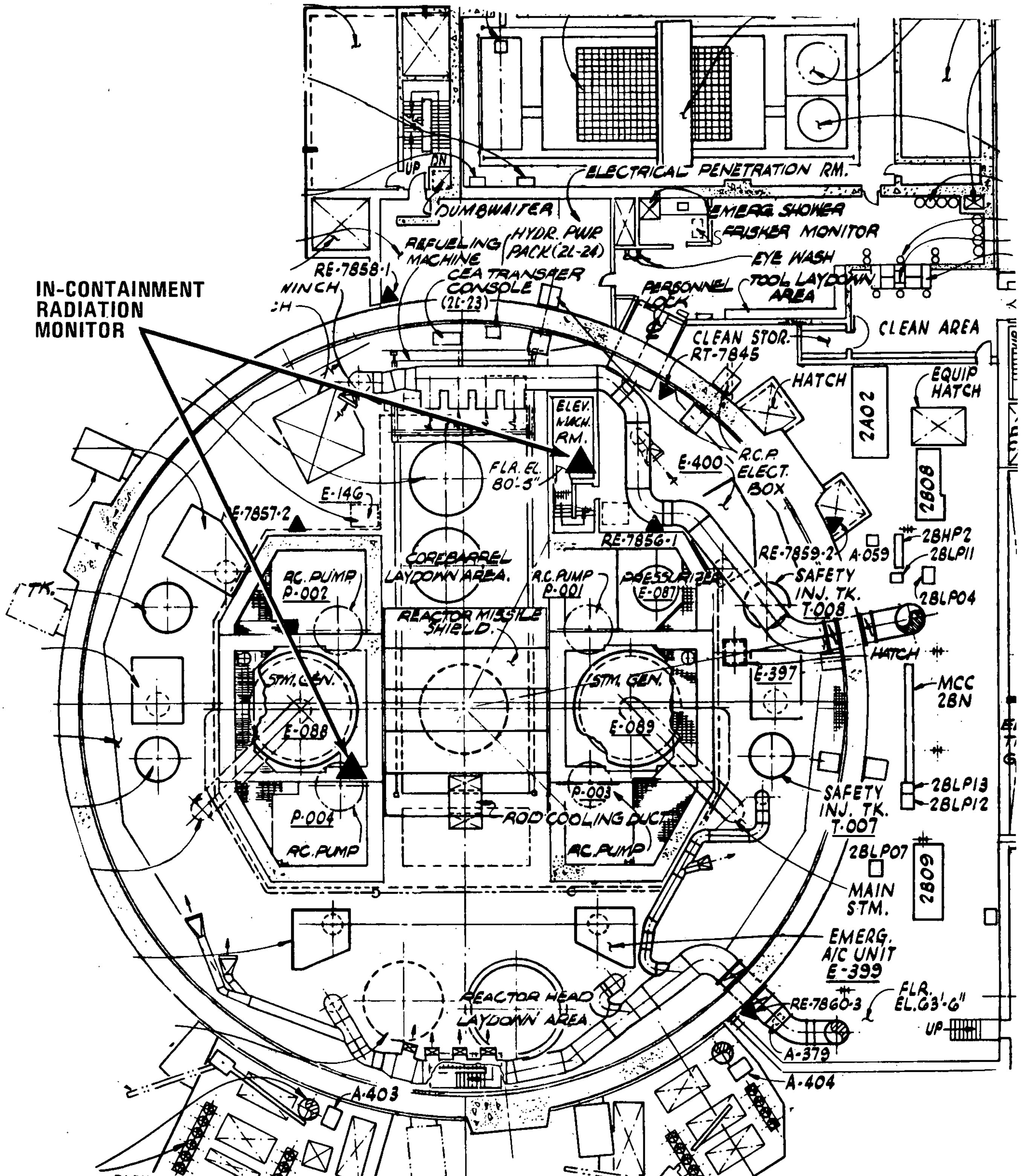


SECTION A

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

MAIN STEAM LINE
MONITOR

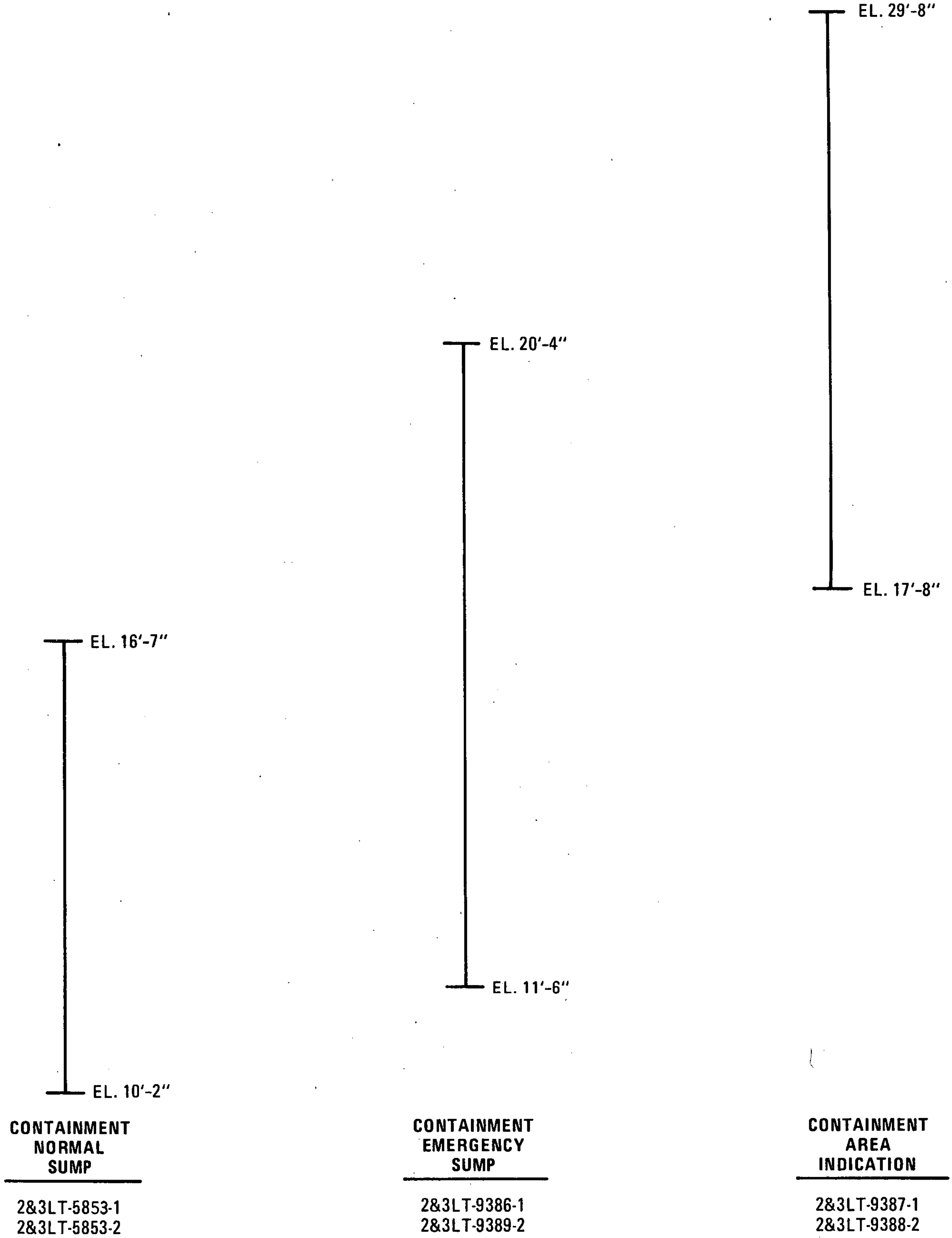
Figure II.F.1-6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

HIGH RANGE RADIATION MONITORS
(UNIT 2 SHOWN)

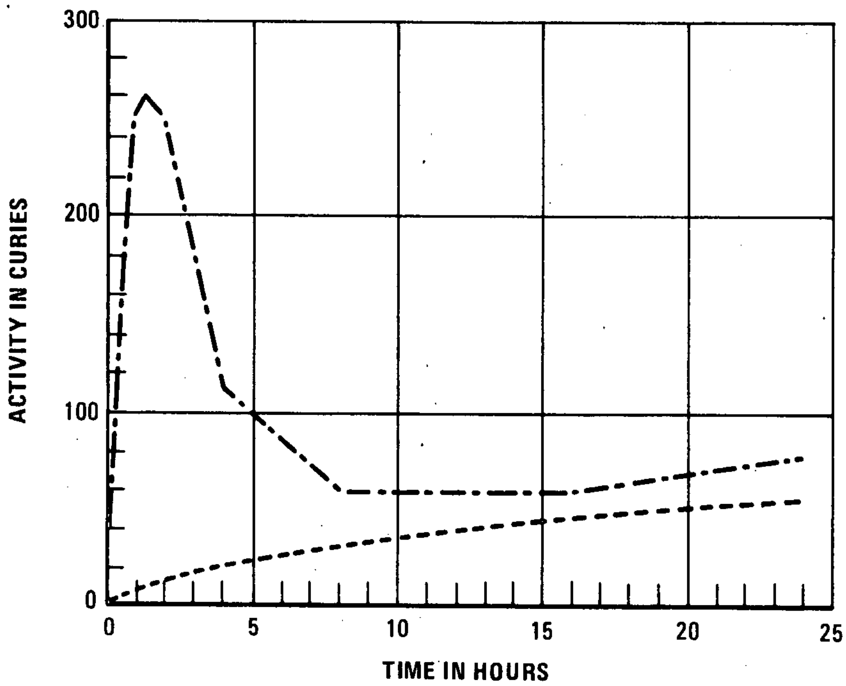
Figure II.F.1-7

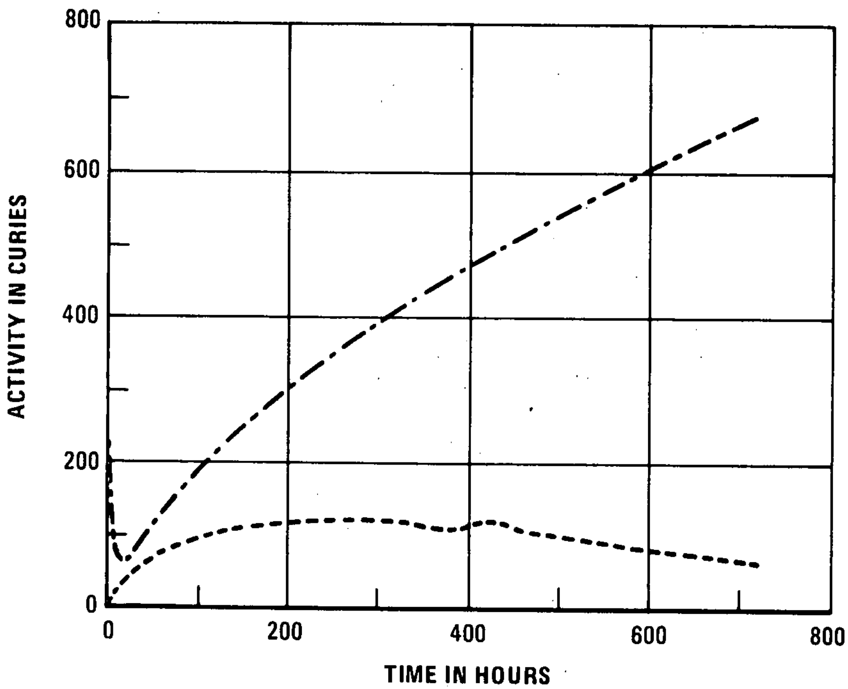


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CONTAINMENT SUMPS AND
AREA INDICATION RANGES

Figure II.F.1-8



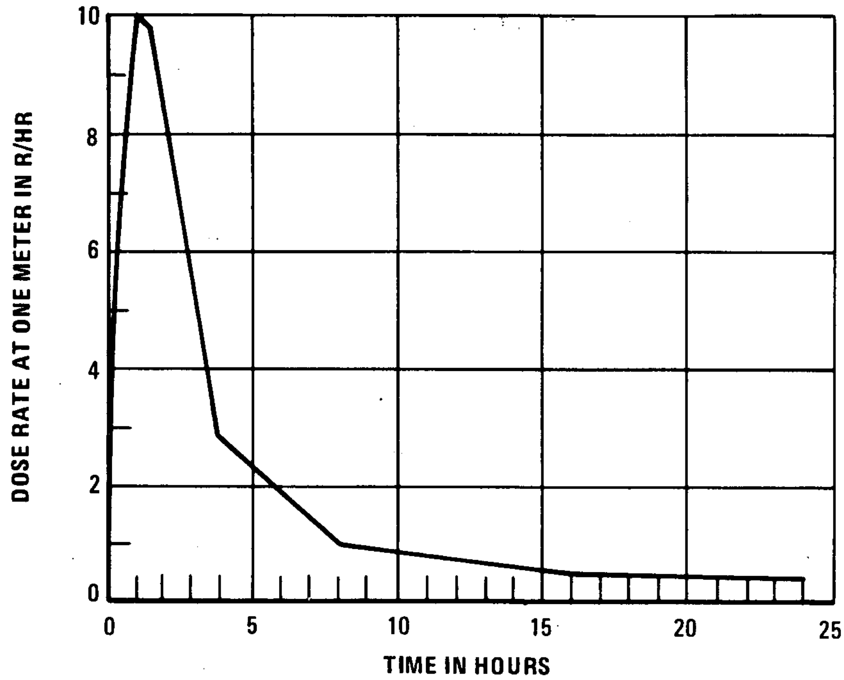


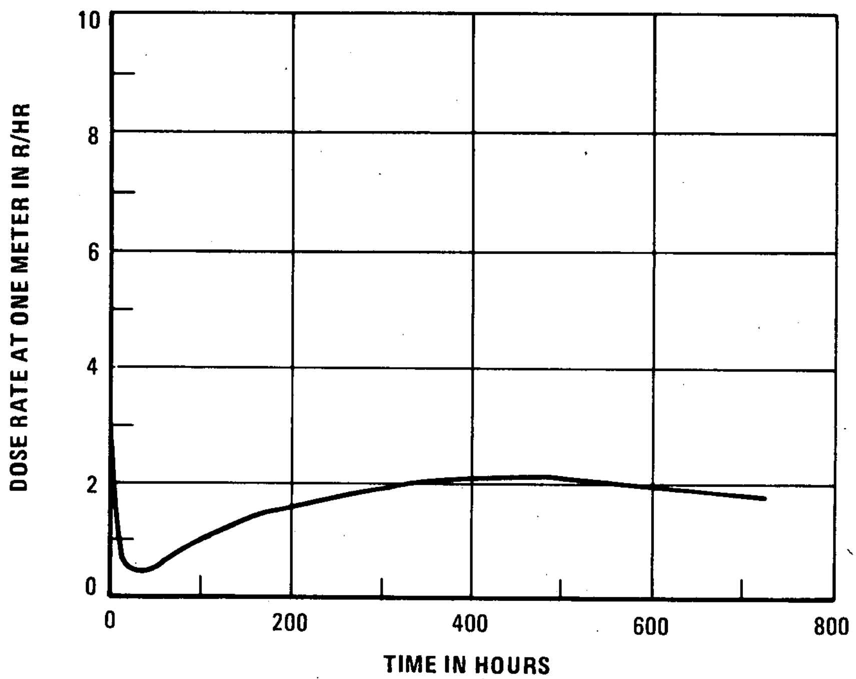
NUCLIDE TYPE

— · — PARTICULATE

------ HALOGENS

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
WIDE RANGE EFFLUENT MONITOR FILTER HALOGEN AND PARTICULATE ACTIVITY
Figure II.F.1-9

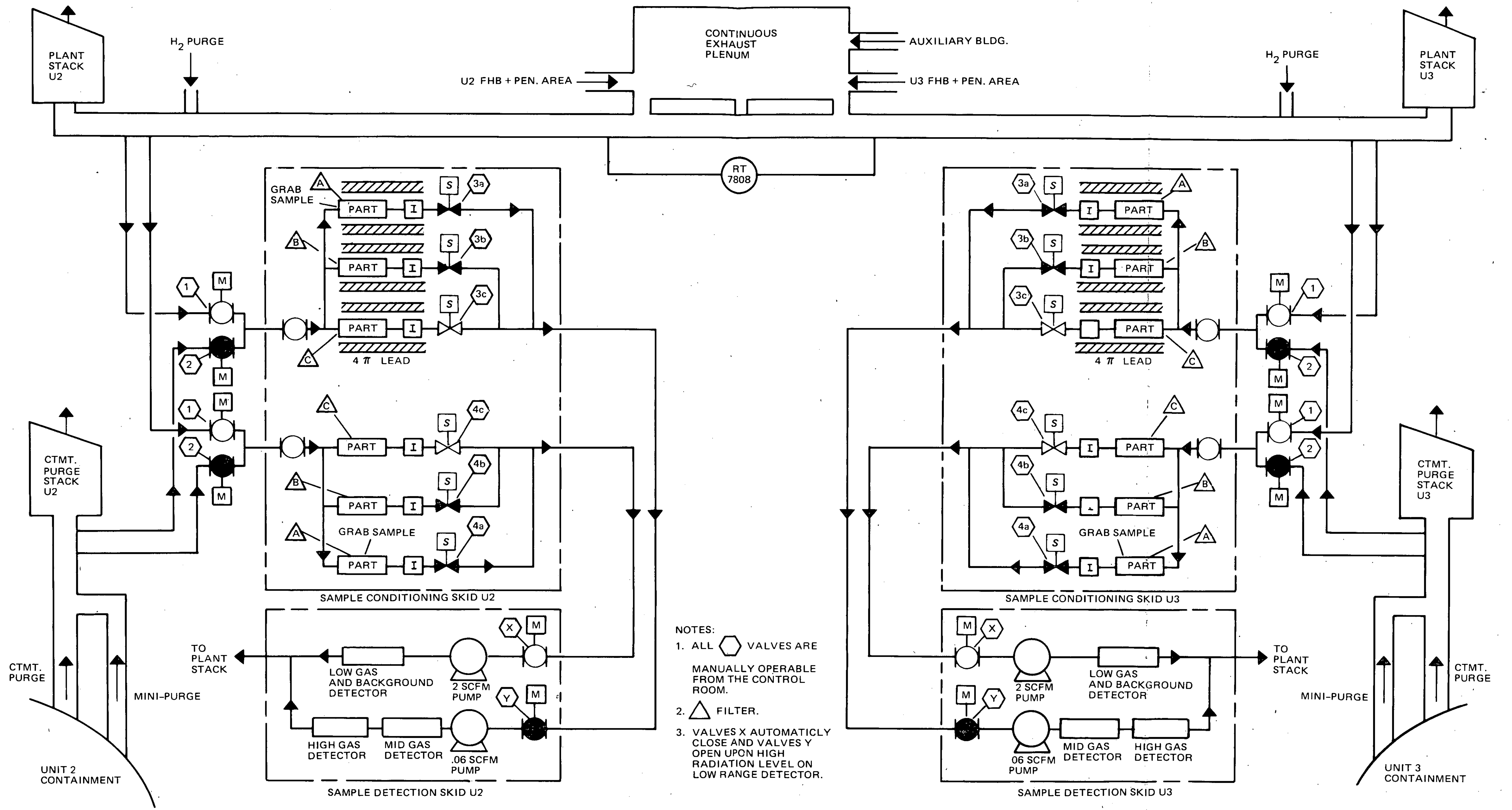




SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

WIDE RANGE EFFLUENT MONITOR
FILTER EXTERNAL DOSE

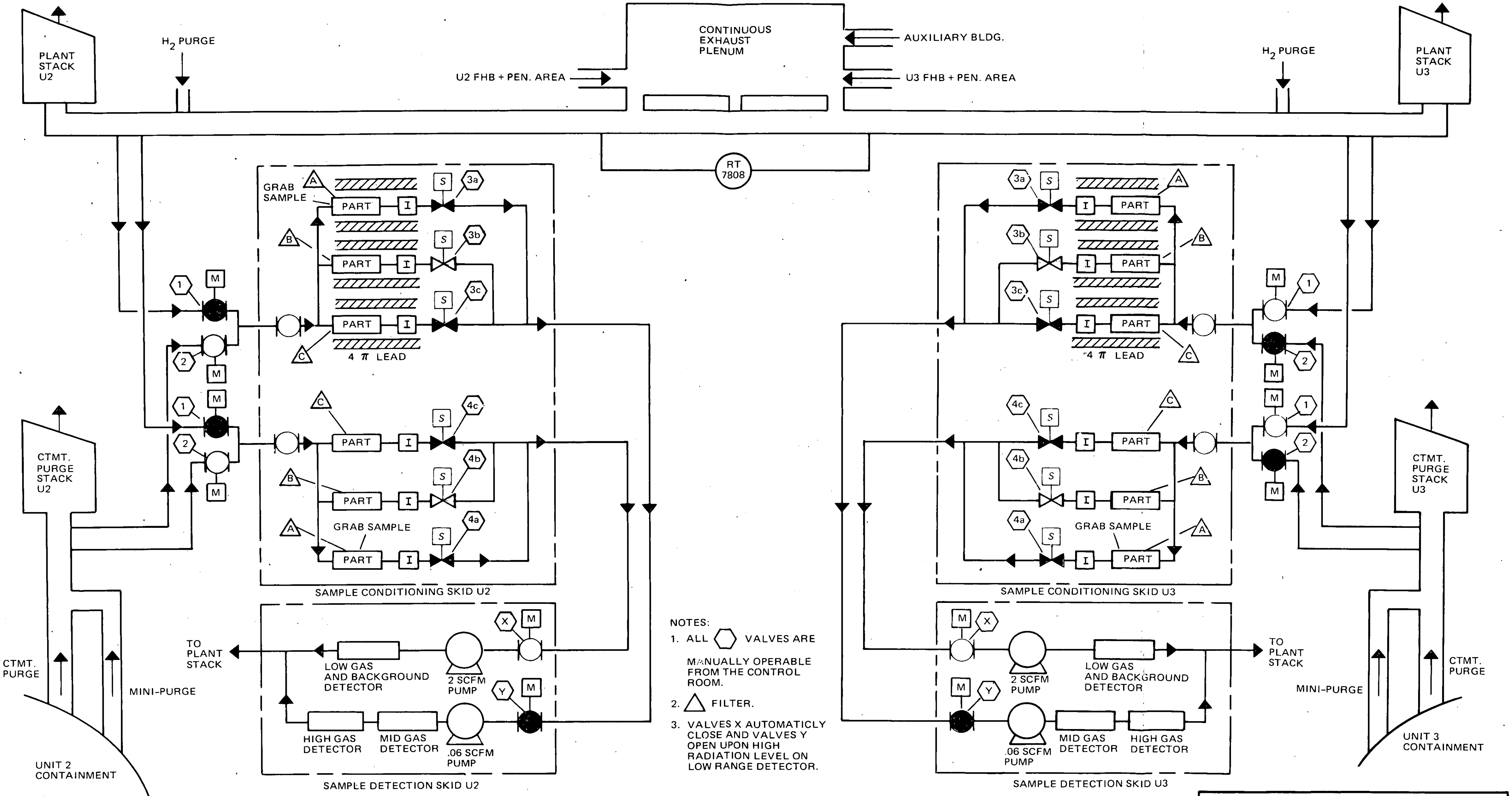
Figure II.F.1-10



SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

WIDE RANGE EFFLUENT MONITOR CONFIGURATION FOR NORMAL OPERATION

Figure II.F.1-11



SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

WIDE RANGE EFFLUENT MONITOR CONFIGURATION FOR CONTAINMENT PURGE FOLLOWING POSTULATED UNIT 2 ACCIDENT

Figure II.F.1-12

II.F.2 - NUREG 0737 - INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Clarification

(1) Design of new instrumentation should provide an unambiguous indication of ICC. This may require new measurements or a synthesis of existing measurements which meet design criteria (item 7).

(2) The evaluation is to include reactor-water-level indication.

(3) Licensees and applicants are required to provide the necessary design analysis to support the proposed final instrumentation system for inadequate core cooling and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core-cooling conditions.

(4) The indication of ICC must be unambiguous in that it should have the following properties:

(a) It must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high-void fraction-pumped flow as well as stagnant boil-off); and,

(b) It must not erroneously indicate ICC because of the presence of an unrelated phenomenon.

(5) The indication must give advanced warning of the approach of ICC.

(6) The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advanced warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided that the indicated temperatures can be correlated to provide indication of the existence of ICC and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit thermocouples to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.

(7) All instrumentation in the final ICC system must be evaluated for conformance to Appendix A, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.

(8) If a computer is provided to process liquid-level signals for display, seismic qualification is not required for the computer and associated hardware beyond the isolator or input buffer at a location accessible for maintenance following an accident. The single-failure criteria of item 2, Appendix A, need not apply to the channel beyond the isolation device if it is designed to provide 99% availability with respect to functional capability for liquid-level display. The display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source which is battery backed. The quality assurance provisions cited in Appendix A, item 5, need not apply to this portion of the instrumentation system. This is a new requirement.

(9) Incore thermocouples located at the core exit or at discrete axial levels of the ICC monitoring system and which are part of the monitoring system should be evaluated for conformity with Attachment 1, "Design and Qualification Criteria for PWR Incore Thermocouples," which is a new requirement.

(10) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

I 2- NUREG 0737- ATTACHMENT 1, DESIGN AND QUALIFICATION CRITERIA FOR PRESSURIZED-WATER REACTOR INCORE THERMOCOUPLES

(1) Thermocouples located at the core exit for each core quadrant, in conjunction with core inlet temperature data, shall be of sufficient number to provide indication of radial distribution of the coolant enthalpy (temperature) rise across representative regions of the core. Power distribution symmetry should be considered when determining the specific number and location of thermocouples to be provided for diagnosis of local core problems.

(2) There should be a primary operator display (or displays) having the capabilities which follow:

(a) A spatially oriented core map available on demand indicating the temperature or temperature difference across the core at each core exit thermocouple location.

(b) A selective reading of core exit temperature, continuous on demand, which is consistent with parameters pertinent to operator actions in connecting with plant-specific inadequate core cooling procedures. For example, the action requirement and the displayed temperature might be either the highest of all operable thermocouples or the average of five highest thermocouples.

(c) Direct readout and hard-copy capability should be available for <u>all</u> thermocouple temperatures. The range should extend from 200°F (or less) to 1800°F (or more).

(d) Trend capability showing the temperature-time history of representative core exit temperature values should be available on demand.

(e) Appropriate alarm capability should be provided consistent with operator procedure requirements.

(f) The operator-display device interface shall be human-factor designed to provide rapid access to requested displays.

(3) A backup display (or displays) should be provided with the capability for selective reading of a minimum of 16 operable thermocouples, 4 from each core quadrant, all within a time interval no greater than 6 minutes. The range should extend from 200°F (or less) to 2300°F (or more).

(4) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions.

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

(5) The instrumentation must be evaluated for conformance to Appendix A, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as modified by the provisions of items 6 through 9 which follow.

(6) The primary and backup display channels should be electrically independent, energized from independent station Class IE power sources, and physically separated in accordance with Regulatory Guide 1.75 up to and including any isolation device. The primary display and associated hardware beyond the isolation device need not be Class IE, but should be energized from a high-reliability power source, battery backed, where momentary interruption is not tolerable. The backup display and associated hardware should be Class IE.

(7) The instrumentation should be environmentally qualified as described in Appendix A, item 1, except that seismic qualification is not required for the primary display and associated hardware beyond the isolater/input buffer at a location accessible for maintenance following an accident.

(8) The primary and backup display channels should be design to provide 99% availability for each channel with respect to functional capability to display a minimum of four thermocouples per core quadrant. The availability shall be addressed in technical specifications.

(9) The quality assurance provisions cited in Appendix A, item 5, should be applied except for the primary display and associated hardware beyond the isolation device.

RESPONSE

Since early 1980 the C-E Owners Group (in Which SCE has been actively participating has conducted an evaluation of response characteristics of instrumentation under conditions of inadequate core cooling. An outline of this evaluation was discussed with the NRC staff at a meeting in Bethesda, MD on May 28, 1980. The instruments whose response characteristics have been evaluated are the subcooled margin monitor, the heated junction thermocouple system, core exit thermocouples, in-core thermocouples, self-powered neutron detectors, hot leg resistance temperature detectors, and ex-core neutron detectors. This evaluation was completed in December, 1980, and the results have recently been distributed to members of the C-E Owners Group. SCE is currently studying these results.

SCE has participated over the past year in the C-E Owners Group development of a technique for measurement of water level in the reactor vessel. The technique which has been selected by the C-E Owners Group is use of heated junction thermocouples (HJTC) distributed at various radial and axial locations in the reactor vessel above the fuel alignment plate. The design objective of this system is to provide a measurement of the water inventory in the reactor vessel above the fuel alignment plate. The details of this design activity were discussed with the NRC staff at a meeting in Bethesda, MD, on May 28, 1980. Since that time, Combustion Engineering has issued a commercial proposal for the HJTC System. SCE is procuring the HJTC System as a component of an instrumentation system for monitoring inadequate core cooling.

Attached is a Summary Status Report on C-E Owners Group inadequate core cooling activities as supplemented by San Onofre specific design/hardware implementation efforts. The Status Report includes a conceptual design description of an Inadequate Core Cooling (ICC) Detection System. SCE believes that this design requires further evaluation and development; however, SCE feels sufficient design and test information has been generated to allow SCE to proceed with procurement of the Subcooled Margin Monitor and Heated junction Thermocouple Systems. In addition, SCE is installing 56 Core Exit Thermocouples (CET) in the in-core instrument assemblies. Continuing evaluation, including the testing program for the HJTC System outlined in the attached report, are being conducted by the C-E Owners Group. SCE is a participating member.

SCE is participating as a member of the C-E Owners Group to define appropriate generic and plant-specific documentation of the ICC Detection System. It is expected that this definition will be completed during the first quarter of 1981. Following completion of this C-E Owners Group effort, SCE will develop a plant-specific definition and schedule for further ICC Detection System documentation.

SAN ONOFRE UNITS 2 AND 3

INADEQUATE CORE COOLING DETECTION SYSTEM

SUMMARY STATUS REPORT

JANUARY 1981

II.F.2.1-2

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 SUMMARY OF ACTIVITIES

This report responds to the requirements in Section II.F.2 of NUREG-0737 (Ref. 1). The report describes the status of design and development activities being conducted by the C-E Owners Group to define a system of instrumentation to be used to detect inadequate core cooling (ICC). The report also provides information specific to San Onofre 2/3 in order to demonstrate the applicability of the generic activity to San Onofre 2/3.

Results of initial studies by the C-E Owners Group are considered documented in reports CEN-117 (Ref. 2) and CEN-125 (Ref. 3). These results were ~~referenced~~ in ~~a letter from~~ the preparation of the four emergency operating instructions which ~~SCE~~ ~~dated~~ ______. All studies have been based on the requirement to indicate the approach to, the existence of and the recovery from ICC.

A three step process is being used to define the ICC Detection System. First, a definition for the state of ICC has been selected. Second, typical accident events which progress toward the defined state of ICC have been analyzed. Third, instruments which indicate the progression of these events have been selected and evaluated.

Based on initial evaluations of a variety of instruments, an ICC Detection System has been defined. This system is judged to be technically sufficient for ICC detection. However, this system is not uniquely necessary, and functions of its various components may be performed by alternative components. This system is described in Section 3. Further developments are necessary before the system can be implemented and these are planned as described throughout this report.

1.2 DEFINITION OF ICC

The definition of ICC and the functional requirements for the ICC Detection System have been established within the bounds of the following core conditions:

SCE transmitted to the NRC by letter dated November 12, 1980.

1. The reactor is tripped so only decay power is considered.

2. The coolant level falls below the top of the core, which can occur only with a loss of coolant mass from the Reactor Coolant System (RCS).

3. The event proceeds slowly enough so that the operator has time to observe and to make use of the instrument displays.

These conditions provide the boundaries for a range of sizes of small break loss of coolant accident (LOCA) caused by either RCS ruptures or primary coolant expansion.

The following definitions of ICC have been considered:

1. First occurrence of saturation.

2. Core uncovery.

3. Fuel clad temperature of 900°F (below which return to normal operation may be permissible).

4. Fuel clad temperature of 1100°F (below which clad rupture in not expected to occur).

5. Fuel clad temperature of 2200°F (which is the licensing limit for design basis events using approved analytical models).

It has been concluded that the events can progress too rapidly for the instrumentation to reliably display the approach of ICC if one of the first four definitions is used. Therefore, it is concluded that definition 5, a fuel clad temperature of 2200°F, should be selected as the criterion for existence of ICC.

1.3 DESCRIPTION OF EVENT PROGRESSION

A typical small break LOCA illustrates the progression of an event which causes the approach to ICC. Figure 1-1 shows a representative behavior for the two phase mixture level and the RCS pressure vs. time for the event. The event progression is divided into four intervals which are shown in Figure 1-1 and are defined in Table 1-1.

1.4 SUMMARY OF SENSOR EVALUATIONS

Several sensors have been evaluated for use in an ICC Detection System. The instruments considered are listed in Table 1-2, where their capabilities are summarized. Significant conclusions about each instrument are given below.

1.4.1 Subcooled Margin Monitor

The Subcooled Margin Monitor (SMM), using input from existing Resistance Temperature Detectors (RTD) in the hot and cold legs and from the pressurizer pressure sensors, is adequate to detect the initial occurrence of saturation during LOCA events and during loss of heat sink events.

The usefulness of the SMM can be significantly increased by also feeding into it the signals from the fluid temperature measurements from the Reactor Vessel Level Monitoring System (RVLMS) and the signals from selected core exit thermocouples and by modifying the SMM to calculate and display degrees superheat (up to about 1800^{o}F) in addition to degrees subcooling. The signals from the RVLMS temperature measurements provide information about possible local differences in temperature between the reactor vessel upper head/upper plenum (location of the RVLMS) and the hot or cold legs (location of the RTDs). The core exit thermocouples respond to the coolant temperature at the core exit and their signal indicates superheat after the coolant level drops below the top of the core and, thus, provide an approximate indication of the depth of core uncovery.

With these modifications, the SMM can be used for detection of the approach to ICC, namely Interval 1 (loss of subcooling), Interval 3 (core uncovery) and Interval 4 (core recovery). Even with the modifications, the SMM will not be capable of indicating the existence of Interval 2 when the coolant is at saturation conditions and the level is between the top of the vessel and the top of the core.

The recovery interval may occur at low system pressure and temperature. Since the errors in the existing SMM calculations increase with lower temperature and pressure, required subcooling margins need to be revised for this situation.

1.4.2 Resistance Temperature Detectors (RTD)

The RTD are adequate for sensing the initial occurrence of saturation. The hot leg RTD range is sufficient to sense saturation for events initiated at power. The cold leg RTD, which have a wider range, are sufficient to sense saturation for events initiated from zero power or shutdown conditions.

The RTD range is not adequate for ICC indications during core uncovery. For depressurization LOCA events, the core may uncover at low pressure, when the saturation temperature is below the lower limit of the hot leg RTD. Initial superheat of the steam will therefore not be detected by the hot leg RTD. As the uncovery proceeds, the superheated steam temperature may quickly exceed the upper limit of the RTD range. In order to be useful during the core uncovery interval, the range of the RTD needs to be increased to cover a temperature range from 100°F to 1800°F.

1.4.3 Reactor Vessel Level Monitoring System

The Reactor Vessel Level Monitoring System (RVLMS) is being designed to show the liquid inventory of the mixture of liquid and vapor coolant above the core. It is an instrument which shows the approach to ICC and is the only one which functions in Interval 2, namely the period from the initial occurrence of saturation conditions until the start of core uncovery.

1.4.4 Core Exit Thermocouples

The core exit thermocouples are adequate to show the approach to ICC after core uncovery for the events analyzed provided that the signal processing and display does not add substantial time delay to the thermal delay at the thermocouple junction. As mentioned above, the core exit thermocouples respond to the coolant temperature at the core exit and indicate superheat after the core is no longer completely covered by coolant. Except for a time delay of about 200 to 400 sec, depending on event, the trend of the change in superheat corresponds to the trend of core uncovery as well as to the accompanying trend of the change in cladding temperature.

1.4.5 Self Powered Neutron Detectors (SPND)

The SPND yield a signal caused by high temperature as the two-phase level falls below the elevation of the SPND. However, testing is required to identify the phenomena responsible for the anomalous behavior of the SPND at TMI-2. At the present, their use is limited to low temperature events (less than 1000°F clad temperature) or to only the initial uncovery portion of an event.

1.4.6 Ex-Core Neutron Detectors

Existing source range neutron detectors are sensitive enough to respond to the formation of coolant voids within the vessel during the events analyzed. However, the signal magnitude is ambiguous because of the effects of varying boron concentration and deuterium concentration in the reactor coolant.

A stack of ex-core detectors gives less ambiguous information on voids and level in the vessel. The relative shape of the axial distribution of signals from a stack of five detectors shows promise as an ICC indicator, but additional development is needed.

1.4.7 In-Core Thermocouples

It appears in general feasible that in-core thermocouples may be added to or substituted for some SPND in the in-core instrument string. They respond more quickly to core uncovery than the core-exit thermocouples. Also, due to thermal radiation from the fuel rods they see temperatures closer to the cladding temperatures than to the steam temperature seen by the core exit thermocouples.

For top mounted in-core instrumentation, the core exit thermocouples may survive longer for deep uncovery events because the thermocouples and their leads see only core exit steam temperature which is less than the fuel clad temperature. For bottom mounted in-core instrumentation, those in-core thermocouples which are located below the two-phase level will survive longer than the core exit thermocouples because the core exit thermocouple leads past down through the high temperature region during core uncovery.

Using a synthesis approach, similar to the one described for core exit thermocouples, it is expected that the in-core thermocouple temperature can be related more directly to the adjacent fuel clad temperature than is possible with a synthesis which uses core-exit thermocouples. However, additional work is required to study the temperature response of in-core thermocouples as well as to develop the mechanical design for incorporating the thermocouples into the in-core instrument string and to develop a synthesis method for calculating fuel cladding temperatures.

Table 1-1

Definition of Intervals in ICC Event Progression

Interval No.	ICC Phase	Bounding Parameter	Description
1	Approach to	Reduction in RCS subcooling until saturation occurs.	Depressurization of RCS to saturation pressure at hot leg temperature or heatup to saturation temperature at safety valve pressure.
2	Approach to	Falling twophase mixture level in upper plenum, down to top of active fuel.	Net loss of coolant mass from RCS accompanied by boiling from continued depressurization and/or decay power.
3	Approach to and/or Existence of	Two phase level falls from top of active fuel until minimum level during event progression occurs or until 2200°F clad temperature occurs.	Two phase level drops in core causing clad heatup and producing superheated steam at core exit.
4	Recovery from	Two phase level rises above top of core.	Coolant addition by ECCS raises level and quenches fuel. ICC progression is defined to terminate when vessel is full or when stable, controllable conditions exist.

II.F.2.1-10

Instruments Included in Evaluations
For ICC Instrumentation System
Table 1-2

Instruments	Development Status	Post-Accident Qualification Status	Indication Provided by Instrument	Non-Ambiguity of Signal	Portion of ICC Event Indicated
Subcooled Margin Monitor	Exists	Qualified	Degree of Subcooling in RCS	Good	Approach
Reactor Vessel Level Monitor	Under Development	Will be Qualified	1) Liquid inventory in upper load 2) Liquid inventory in upper plenum 3) Axial temperature distribution in head and plenum	Good Good Good	Approach Recovery
Core Exit Thermocouples	Exit	Can be done	1) Liquid temperature at core exist 2) Infer with synthesis a) Calculated mixture level in-core b) Calculated Clad temperature	Good Fair Uncertain	Approach Existence Recovery
In-Core Thermocouples	Concept Stage	Can be done	1) Metal temperature inside guide tube when RCP off 2) Infer: a) Effective moxture level in-core b) Clad temperature	Good Uncertain	Approach Existence Recovery
Self Powered Neutron Detectors	Exist	Can be done	Indirect measure of mixture level (Low-pressure uncovery)	Poor	Approach Only
Hot Leg RTD (5 each)	Exit	Qualified	Fluid temperature in hot leg Infer calculated mixture level and Clad temperature	Good Fair Uncertain	Approach Existence Recovery
Ex-Core Neutron Detector (One, Source range)	Exist	Can be done	Indirect measure of gross voiding Indirect indication of mixture level in-core RCP off	Fair	Approach Existence Recovery
Ex-Core Neutron Detector (Stack of 5, Source range)	Concept	Can be done	Same as one ex-core detector, but more axial resolution	Fair	Approach Existence Recovery

II.F.2.1-11

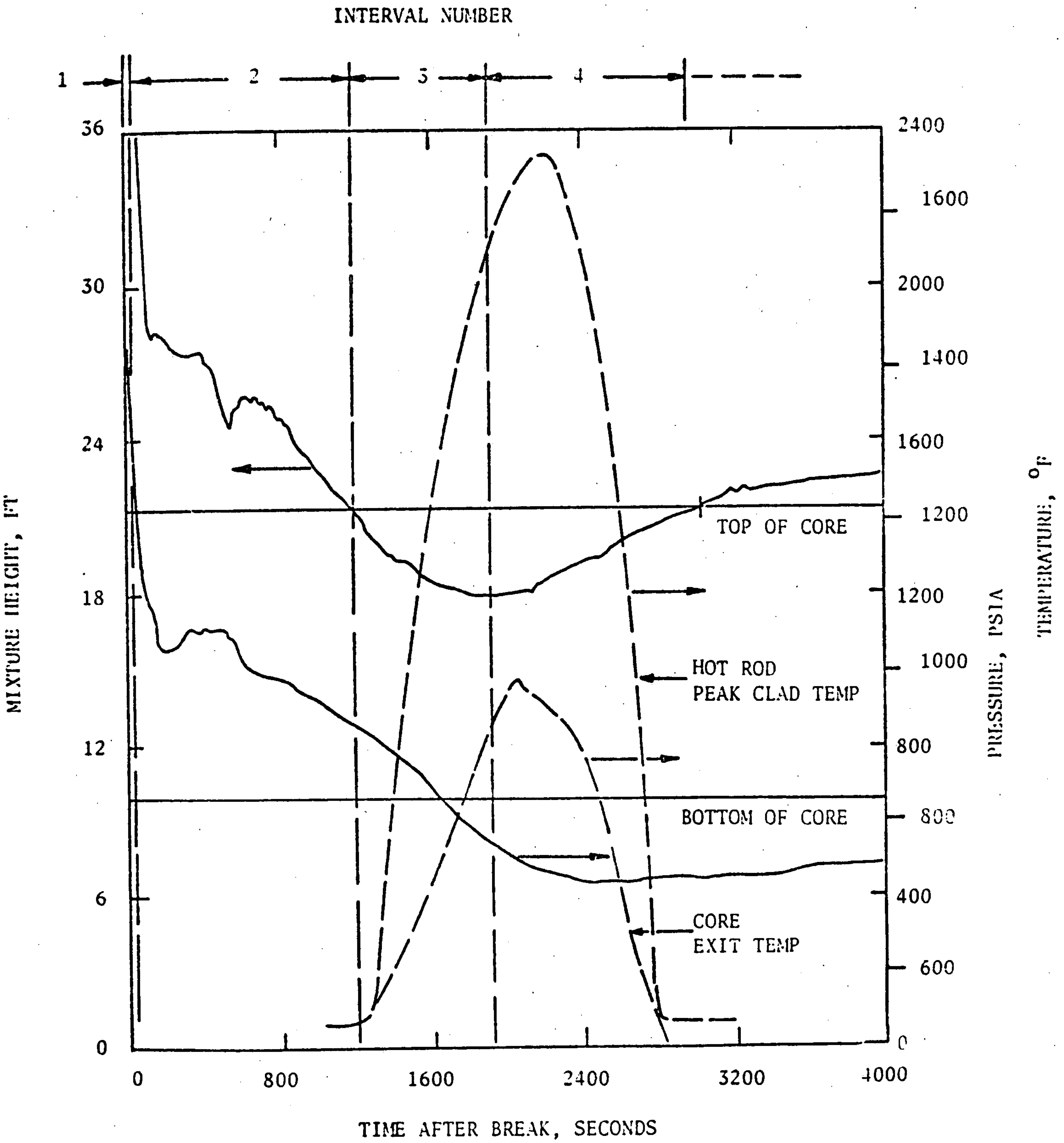


FIGURE 1-1

DEFINITION OF INTERVALS IN EVENT PROGRESSION

2.0 SYSTEM FUNCTIONAL DESCRIPTION

In the following sections a functional description of the instruments of the ICC Detection System is given and the function of the instruments is related to the ICC intervals which were described in Section 1.0.

2.1 SUBCOOLING AND SATURATION

The parameters measured to detect subcooling and saturation are the RCS coolant temperature and the pressurizer pressure. Temperature is measured in the hot legs for typical LOCA type events and is measured in the vessel upper head region for cooldown events. The measurement range extends from the shutdown cooling conditions up to saturation conditions at the pressurizer safety valve setpoint. The response time needs to be such that the operator obtains adequate information during those events which proceed slowly enough for him to observe and to act upon the information from the instrument display. Plant specific analyses for San Onofre will be performed for a selection of small break LOCA events in order to establish the required response times. Generic analyses done to date show that existing or planned instruments have adequate range and response.

The information which is derived from the reactor vessel temperature and pressure measurements is the amount of subcooling during the initial approach to saturation conditions and the occurrence of saturation during Interval 1. During Interval 4, the reestablishment of subcooled conditions is obtained.

2.2 COOLANT LEVEL MEASUREMENT IN REACTOR VESSEL

The Reactor Coolant System is at saturation conditions until sufficient coolant is lost to lower the two-phase level to the top of the active core. During this interval there are no existing instruments which would measure directly the coolant inventory loss. A Reactor Vessel Level Monitoring System provides direct measurement during this period. The parameter which is measured is the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass which is in the reactor vessel above the core. Measurement of the collapsed water level was selected in preference to measuring two-phase level, because it is a direct indication of the water inventory while the two phase level is determined by water inventory and void fraction.

The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it is intended to function during Interval 4, the recovery interval. Therefore it must survive the high steam temperature which may occur during the preceeding core uncovery interval.

The level range extends from the top of the vessel down to the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression during Intervals 2 and 4 and to detect the consequences of his mitigating actions or the functionability of automatic equipment.

2.3 FUEL CLADDING HEATUP

The overall intent of ICC detection is understood to be the detection of the potential for fission product release from the reactor fuel. The parameter which is most directly related to the potential for fission product release is the cladding temperature rather than the uncovery of the core by coolant.

Since clad temperature is not directly measured, a parameter to which cladding temperature may be related is measured. This parameter is the fluid temperature at the core exit. After the core becomes uncovered; the fluid leaving the core is superheated steam and the amount of superheat is related to the fuel length exposed and to the cladding temperature.

The amount of superheat of the steam leaving the core will be measured by the core exit thermocouples. The time behavior of the superheat temperature is, with the exception of an acceptably small time delay, similar to the time behavior of the cladding temperature. Thus, from the observation of the steam superheat, the behavior of the cladding temperature can be inferred. Observation of the cladding temperature trends during an accident is considered to be of more value to the operator than information on the absolute value of the cladding temperature.

The core exit steam temperature is measured with the thermocouples included in the In-Core Instrument (ICI) string. They are located inside the ICI support tube, at an elevation a few inches above the fuel alignment plate. Generic calculations of a similar installation for representative uncovery events show that the thermocouples respond sufficiently fast to the increasing steam temperature. Plant specific calculations on the San Onofre configuration will be made to verify this response.

The required temperature range of the thermocouples extends from the lowest saturation temperature at which uncovery may occur up to the maximum core average exit temperature which occurs when the peak clad temperature reaches 2200°F. The required thermocouple range is therefore 200°F to about 1800°F, which is the approximate upper service temperature limit. Thermocouples are expected to function with reduced accuracy at even higher temperatures, so the range for processing the thermocouple output extends to about 2300°F.

3.0 SYSTEM CONCEPTUAL DESIGN DESCRIPTION

The following sensors have been selected by SCE as the basic instruments to meet the functional requirements described in Section 2.

1. the Subcooled Margin Monitor (SMM) (Ref. 1),
2. the Heated Junction Thermocouple (HJTC) System (Ref. 2), and
3. the Core Exit Thermocouple (CET) System.

The conceptual design of each ICC instrument is described in this section which addresses:

1. sensors design
2. signal processing design
3. display design.

Figure 3-1 is a typical functional block diagram for the ICC instrument systems. Each instrument system consists of two safety grade channels from sensors through signal processing equipment. The outputs of processing equipment systems feeding the primary display are isolated to separate safety grade and non safety grade systems. Channelized safety grade backup displays are included for each instrument system. The following sections present details of the conceptual design.

3.1. SENSORS DESIGN

3.1.1 Subcooled Margin Monitoring System

The specific subcooled margin monitor design configuration being implemented by SCE prior to fuel load is detailed in Appendix A. SCE is evaluating the need to expand the capability of the subcooled margin monitor to process HJTC System and core exist thermocouples. This would allow the subcooled margin monitor to identify superheat conditions, i.e., core uncovery.

3.1.2 Heated Junction Thermocouple (HJTC) System

The HJTC System measures reactor coolant liquid inventory with discrete HJTC sensors located at different levels within a separator tube ranging from the top of the core to the reactor vessel head. The basic principle of system operation is the detection of a temperature difference between adjacent heated and unheated thermocouples.

As pictured in Figure 3-2, the HJTC sensor consists of a Chromel-Alumel thermocouple near a heater (or heated junction) and another Chromel-Alumel thermocouple positioned away from the heater (or unheated junction). In a fluid with relatively good heat transfer properties, the temperature difference between the adjacent thermocouples is very small. In a fluid with relatively poor heat transfer properties, the temperature difference between the thermocouples is large.

Two design features ensure proper operation under all thermal-hydraulic conditions. First, each HJTC is shielded to avoid overcooling due to direct water contact during two phase fluid conditions. The HJTC with the splash shield is referred to as the HJTC sensor (See Figure 3-2). Second, a string of HJTC sensors is enclosed in a tube that separates the liquid and gas phases that surround it.

The separator tube creates a collapsed liquid level that the HJTC sensors measure. This collapsed liquid level is directly related to the average liquid fraction of the fluid in the reactor head volume above the fuel alignment plate. This mode of direct in-vessel sensing reduces spurious effects due to pressure, fluid properties, and non-homogeneities of the fluid medium. The string of HJTC sensors and the separator tube is referred to as the HJTC instrument.

The HJTC System is composed of two channels of HJTC instruments. Each HJTC instrument is manufactured into a probe assembly. The probe assembly includes eight (8) HJTC sensors, a seal plug, and electrical connectors (Figure 3-3). The eight (8) HJTC sensors are electrically independent and located at eight levels from the reactor vessel head to the fuel alignment plate.

The probe assembly is housed in a stainless steel structure that protects the sensors from flow loads and serves as the guide path for the sensors. Installation arrangements have been developed For San Onofre Units 2 and 3; details of which are provided in Appendix B.

3.1.3 Core Exit Thermocouples (CET) System

The San Onofre reactor contains 56 thermocouples that are top mounted and placed above the fuel assemblies above the fuel alignment plate. Figure 3-4 shows the CET locations. The thermocouples are Type K (Chromel-Alumel) and are connected in the same Incore Instrumentation (ICI) cabling as the fixed incore neutron detectors. The thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies. The junction of each thermocouple is located above the fuel assembly inside a structure which supports and shields the instrument string from flow forces in the outlet plenum region.

The basic design of the CETs will not change for the ICC Detection System. However, modifications will be made to the design detailed in Appendix C.

to meet the qualification requirements (See Section 5.0). The CETs have a maximum usable temperature range from 200°F to approximately 2300°F (Reference 6).

3.2 SIGNAL PROCESSING EQUIPMENT DESIGN

The processing equipment of the ICC instruments is presently being developed. The processing equipment portion will be composed of a combination of new and existing equipment. The design objective for the equipment is to address the NUREG-0737 criteria, including the criteria of Attachment 1 to II.F.2 and Appendix A. The following description present functional and general hardware design criteria in terms of the three instrument systems described in Section 3.1.

The processing hardware will be configured to provide information to the displays described in Section 3.3. The processing equipment includes operator interfaces for equipment testing, setup, and maintenance. The descriptions are for each of the two separate channels.

All three ICC instrument systems will have similar sensor input processing. The outputs of the sensors will be transmitted to

The processor, all of which is outside of containment, using approximately qualified cable systems. SCE is evaluating the need to upgrade the existing cabling systems for the CETs. This evaluation will be completed by June 1981.

The processing for the ICC instrumentation will have surveillance testing and diagnostic capabilities. Automatic on-line surveillance tests will continuously check for specified hardware and software malfunctions. The on-line automatic surveillance tests as a minimum will indicate inputs that are out of range and computer hardware malfunctions. The malfunctions will be indicated through the operator interface. A manual on-line diagnostic capability will be incorporated to aid the operator in locating the source of these malfunctions.

3.2.1 Subcooled Margin Monitoring System

The SMM processing equipment will perform the following functions:

1. Calculate the subcooled margin.

   The saturation temperature is calculated from the minimum pressure input and the saturation pressure is calculated from the maximum temperature input (See Section 3.1). The temperature subcooled margin is the difference between saturation temperature and the maximum temperature input. The pressure subcooled margin is the difference between saturation pressure and the minimum pressure input.

2. Process all outputs for display.

3. Provide an alarm output when subcooled margin reaches a preselected setpoint.

The SMM will accept the temperature and pressure inputs over the input range of the sensors,

the RTDs from 0°F to 710°F,

and the pressure from 0 psia to 3000 psia. The saturation temperature and pressure are calculated from a saturation curve derived from the 1967 ASME steam tables and an interpolation routine.

3.2.2 HJTC System

The processing equipment for the HJTC performs the following functions:

1. Determine if liquid inventory exists at the HJTC positions.

   The heated and unheated thermocouples in the HJTC are connected in such a way that absolute and differential temperature signals are available. This is shown in Figure 3-5. When water surrounds the thermocouples, their temperature and voltage output are approximately equal. $V_{(A-C)}$ on Figure 3-5 is, therefore, approximately zero. In the absence of liquid, the thermocouple temperatures and output voltages become unequal, causing $V_{(A-C)}$ to rise. When $V_{(A-C)}$ of the individual HJTC rises above a predetermined setpoint, liquid inventory does not exist at this HJTC position.

2. Determine the maximum upper plenum/head fluid temperature from the unheated thermocouples for use as an input to the SMM. (The temperature processing range is from 100°F to 1800°F.)

3. Process all inputs and calculated outputs for display.

4. Provide an alarm output to the plant annunciator system when any of the HJTC detects the absence of liquid level.

The CFM System is a dedicated, computer based display system that monitors critical plant functions:

1. Core reactivity control
2. Core heat removal control
3. RCS inventory control
4. RCS pressure control
5. RCS heat removal control
6. Containment pressure/temperature control
7. Containment isolation

If any of the critical functions are violated, (by exceeding logic setpoints) a Critical Function Alarm (CFA) is initiated. The ICC instruments outputs will be incorporated in this critical function alarm logic.

The CFM displays data on four cathode ray tubes. The data has three levels of information:

Level 1 Critical functions status (very general)
Level 2 System overview (general, on system)
Level 3 System detail (specific information)

This hierarchy allows the operator to progress from an overall system view to a detailed diagnostic view. The ICC instrument outputs will be incorporated in all three levels of display. The detailed ICC information is anticipated to be displayed on the Level 3 display. Trending displays are also available with the CFM.

Each channel of backup display will present the most reliable basic information for each of the ICC instrument systems. These displays will be human engineered to give the operator clear unambiguous indications. The backup displays are designed:

1. to give primary instrument indications in the remote chance that the primary display becomes inoperable.
2. to provide confirmatory indications to the primary display.
3. to aid in surveillance tests and diagnostics.

The following sections present details on the display for each of the instrument systems as presently conceived.

3.3.1 Subcooled Margin Monitor Display

The following information is anticipated to be presented on the primary display:

1. Pressure margin to saturation.
2. Temperature margin to saturation.
3. Maximum temperature and source (i.e., RTD, H/TC or CET (if upgraded).
4. Minimum pressure

The following information is anticipated to be presented on the backup displays:

1. Pressure margin to saturation
2. Temperature margin to saturation
3. Temperature inputs
4. Pressure inputs

5. Provide control of heater power for proper HJTC output signal level. Figure 3-6 shows a single channel conceptual design which includes the heater power controller.

3.2.3 Core Exit Thermocouple System

The processing equipment for the CET will perform the following functions:

1. Process all core exit thermocouple inputs for display. Half of the available CET inputs will be processed in each channel.

2. Provide an alarm output to the plant annunciator system when the temperature from any of the CET's exceeds a preselected setpoint.

3. Determine the maximum CET temperature to be supplied to the SMM. The processed temperature range will be from 100°F to 2300°F.

These functions are intended to meet the design requirements of NUREG-0737, II.F.2 Attachment 1. The current design of the process equipment associated with the CET System will be provided in Appendix C.

3.3 DISPLAY DESIGN

The ICC instrument outputs will be displayed through a human engineered cathode ray tube (CRT) based primary display and separate backup displays. The Critical Function Monitor (CFM) System (See Response to NUREG-0737 item III.A.1.2) (Ref. 7) is being considered as the primary display for the ICC instrument outputs. As shown in Figure 3-1, each channel of the ICC instrument system will also have safety grade backup displays. Both primary and backup displays are intended to be designed consistent with the criteria in NUREG-0737 Action Item II.F.2, II.F.2 Attachment 1, and Appendix A. The following description presents the conceptual design for display.

3.3.2 Heated Junction Thermocouple System Display

The following information is anticipated to be displayed on the primary display:

1. Two channels of 8 discrete HJTC positions indicating liquid inventory above the fuel alignment plate.

2. Maximum unheated junction temperature of each of the two channels which is provided to the SMM.

The following information is anticipated to be displayed on the backup displays:

1. Liquid inventory level above the fuel alignment plate derived from the 8 discrete HJTC positions
2. Unheated junction temperature at the 8 positions
3. Heated junction temperature at the 8 positions

3.3.3 Core Exit Thermocouple System Display

The following information is anticipated to be displayed on the primary display:

1. A spatially oriented core map indicating the temperature at each of the CET locations.

2. A selective reading of CET temperature

   At least the maximum CET temperature of each of the two channels which is provided to the SMM will be presented. The backup displays are anticipated to display at least four CET from each quadrant with an identification number for each CET temperature. At least 16 CET temperature will be displayed within 6 minutes.

II.F.2.1-25

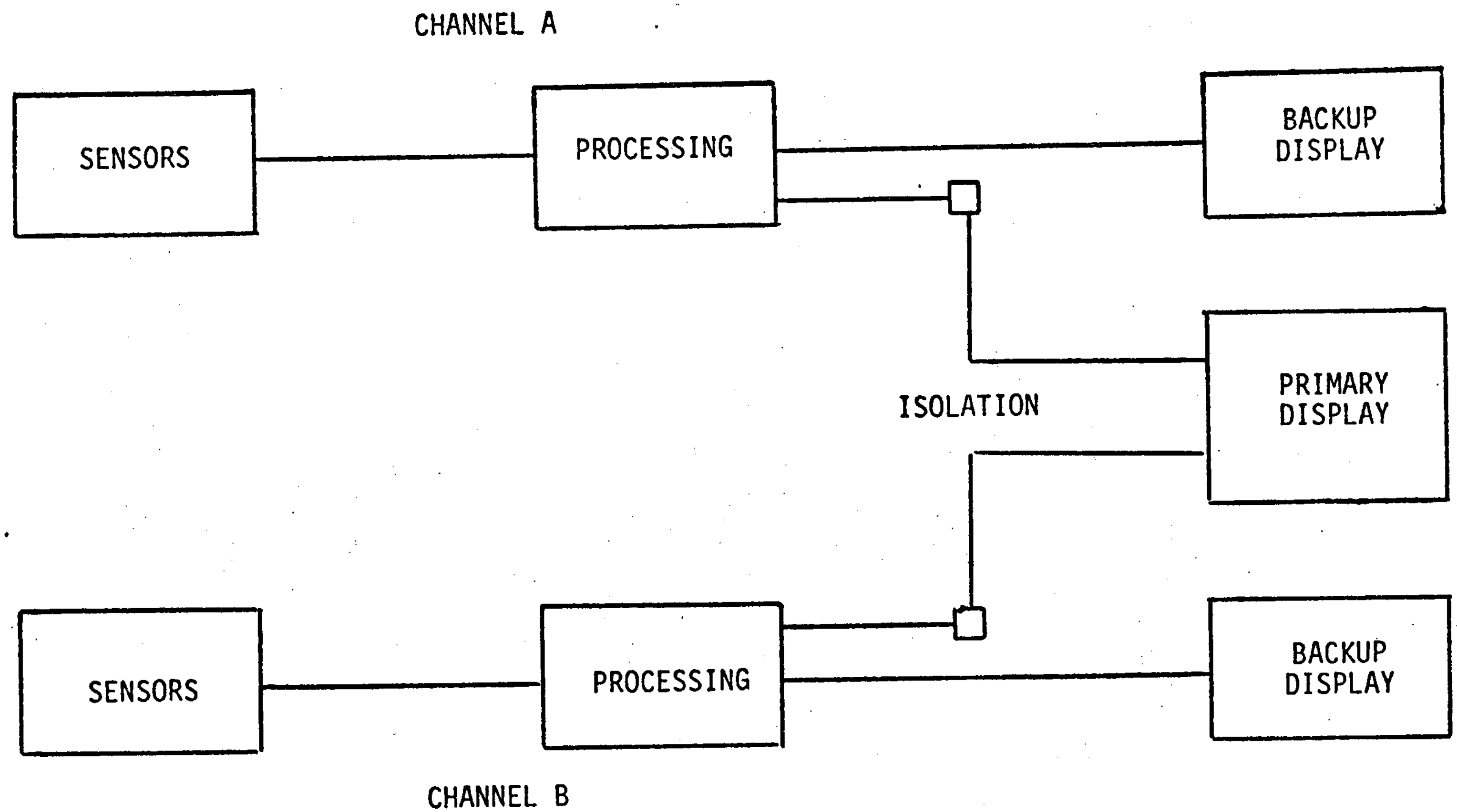


FIGURE 3-1
ICC INSTRUMENTS FUNCTIONAL BLOCK DIAGRAM

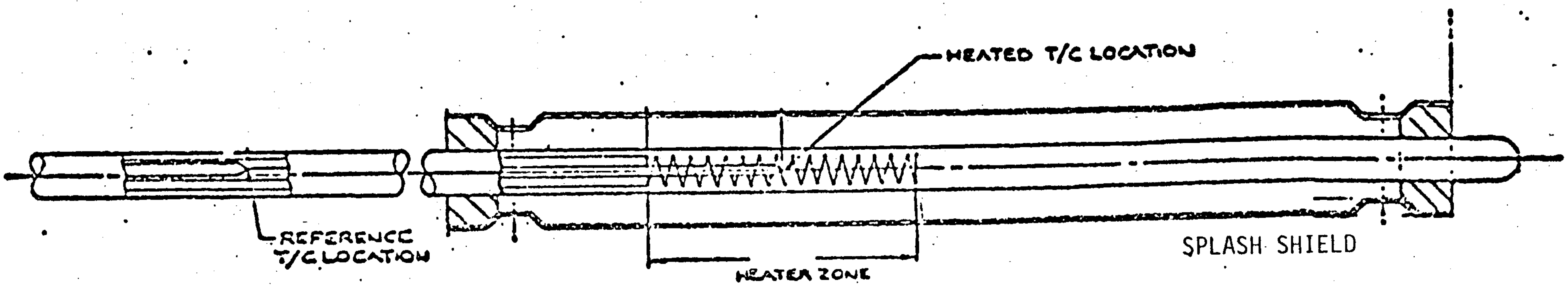


FIGURE 3-2

HJTC SENSOR - HJTC/SPLASH SHIELD

II.F.1-26

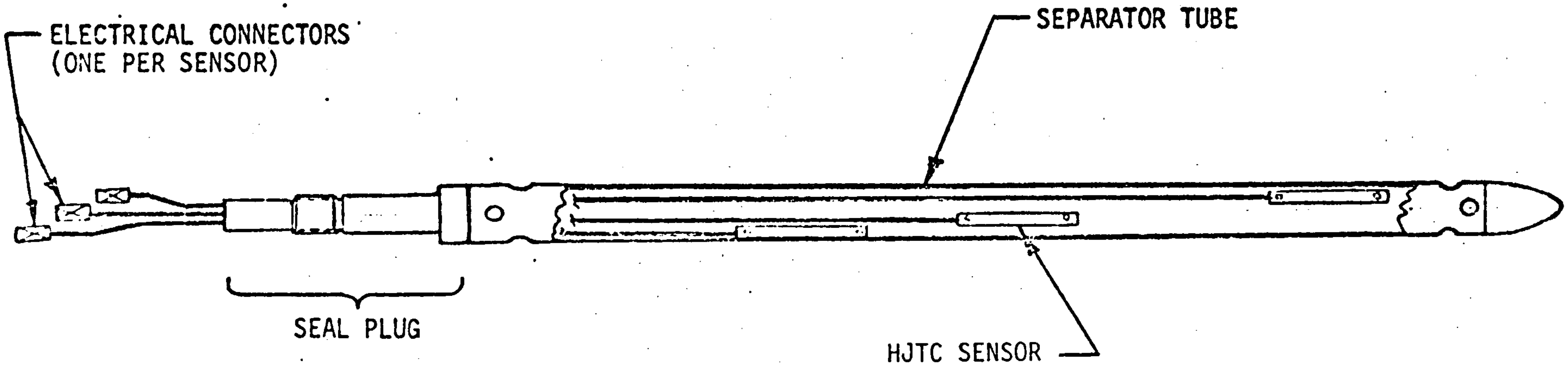


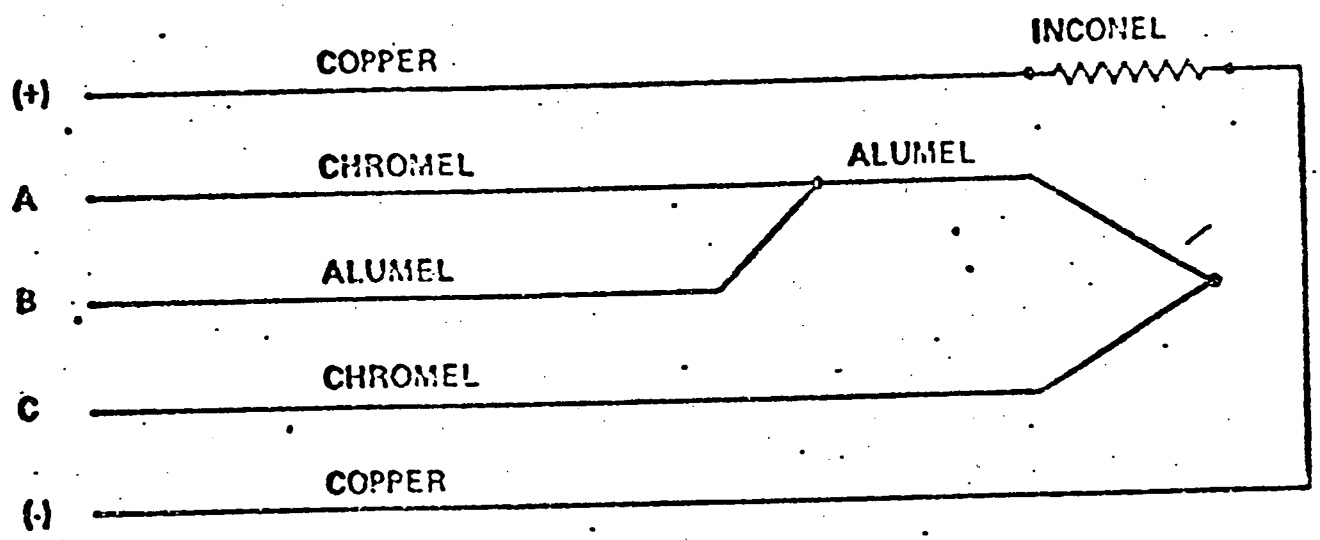
FIGURE 3-3
HEATED JUNCTION THERMOCOUPLE
PROBE ASSEMBLY

II.F.2.1-27

FIGURE 3-4

SAN ONOFRE

FIXED IN-CORE INSTRUMENTATION LOCATIONS



V (A - B) = ACTUAL TEMPERATURE, UNHEATED JUNCTION
V (C - B) = ACTUAL TEMPERATURE, HEATED JUNCTION
V (A - C) = DIFFERENTIAL TEMPERATURE

FIGURE 3-5

ELECTRICAL DIAGRAM OF H.J.T.C.

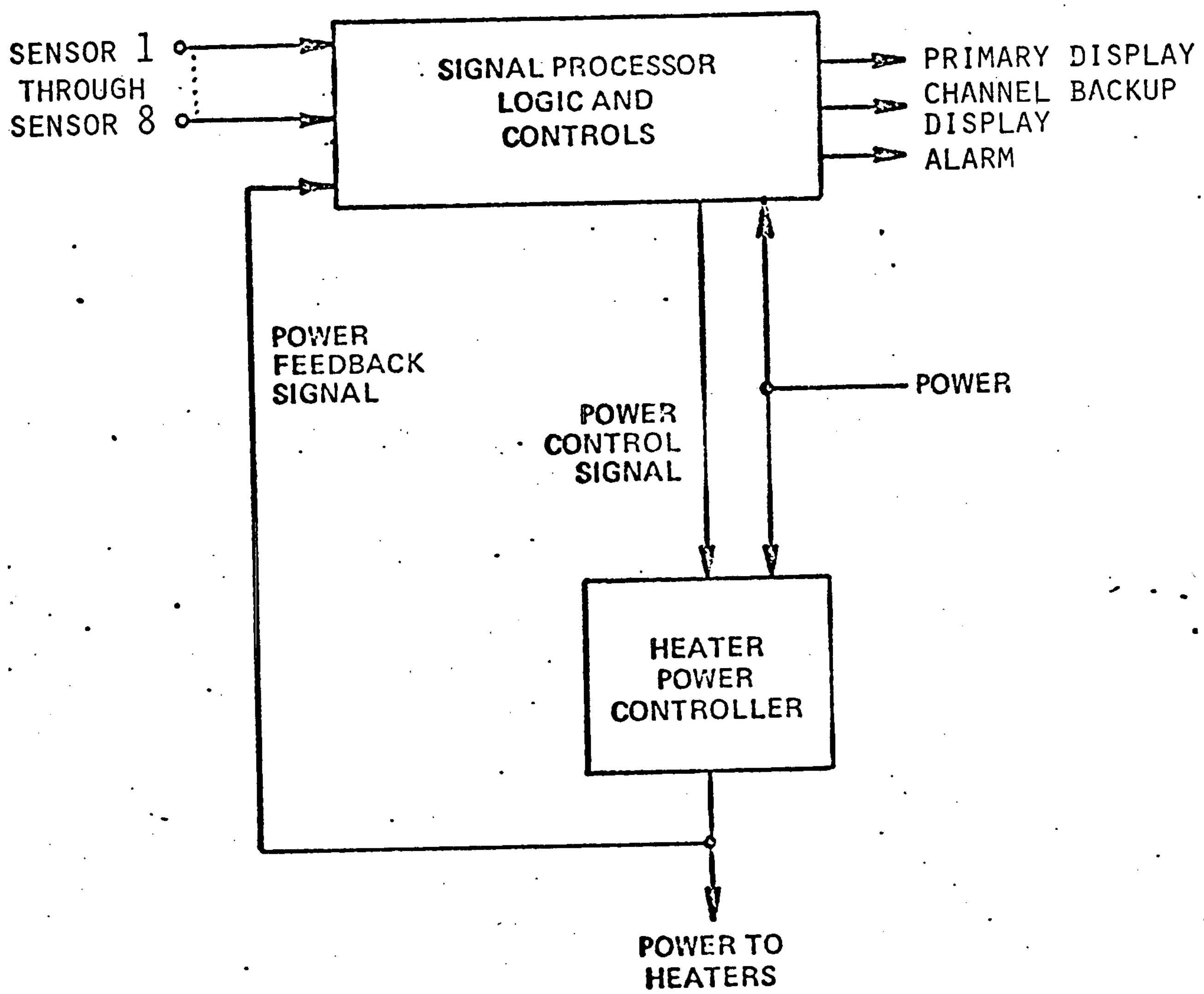


FIGURE 3-6

HJTC SYSTEM PROCESSING CONFIGURATION
(ONE CHANNEL SHOWN)

4.0 SYSTEM VERIFICATION TESTING

This section describes tests and operational experience with ICC system instruments.

4.1 RTD AND PRESSURIZER PRESSURE SENSORS

The hot and cold leg RTD temperature sensors and the pressurizer pressure sensors are standard NSSS instruments which have well known responses. No special verification tests have been performed nor are planned for the future. These sensors provide basic, reliable temperature and pressure inputs which are considered adequate for use in the SMM and other additional display functions.

HJTC SYSTEM SENSORS AND PROCESSING

The HJTC System is a new system developed to indicate liquid inventory above the core. Since it is a new system, extensive testing has been performed and further tests are planned to assure that the HJTC System will operate to unambiguously indicate liquid inventory above the core.

The testing is divided into three phases:

Phase 1 - Proof of Principle Testing
Phase 2 - Design Development Testing
Phase 3 - Prototype Testing

The first phase consisted of a series of five tests, which have been completed. The testing demonstrated the capability of the

HJTC instrument design to measure liquid level in simulated reactor vessel thermal-hydraulic conditions (including accident conditions).

Test 1 Autoclave test to show HJTC (thermocouples only) response to water or steam.

In April 1980, a conceptual test was performed with two thermocouples in one sheath with one thermocouple as a heater and the other thermocouple as the inventory sensor. This configuration was placed in an autoclave (pressure vessel with the capabilities to adjust temperature and pressure). The thermocouples were exposed to water and then steam environments. The results demonstrated a significant output difference between steam and water conditions for a given heater power level.

Test 2 Two phase flow test to show bare HJTC sensitivity to voids.

In June 1980, a HJTC (of the present differential thermocouple design) was placed into the Advanced Instrumentation for Reflood Studies (AIRS) test facility, a low pressure two phase flow test facility at Oak Ridge National Laboratory (ORNL). The HJTC was exposed to void fractions at various heater power levels. The results demonstrated that the bare HJTC output was virtually the same in two phase liquid as in subcooled liquid. The HJTC did generate a significant output in 100% quality steam.

Test 3 Atmospheric air-water test to show the effect of a splash shield

A splash shield was designed to increase the sensitivity to voids. The splash shield prevents direct contact with the liquid in the two phase fluid. The HJTC output changed at intermediate

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Test 3 Atmospheric air-water test to show the effect of a splash shield

A splash shield was designed to increase the sensitivity to voids. The splash shield prevents direct contact with the liquid in the two phase fluid. The HJTC output changed at intermediate void fraction two phase fluid. The results demonstrated that the HJTC sensor (heated junction thermocouple with the splash shield) sensed intermediate void fraction fluid conditions.

Test 4 High pressure boil-off test to show HJTC sensor response to reactor thermal-hydraulic conditions.

In September 1980, a C-E HJTC sensor (HJTC with splash shield) was installed and tested at the ORNL Thermal-Hydraulics Test Facility (THTF). The device is still installed and available for further tests at ORNL. The HJTC sensor was subjected to various two phase fluid conditions at reactor temperatures and pressures. The results verified that the HJTC sensor is a device that can sense liquid inventory under normal and accident reactor vessel high pressure and temperature two phase conditions.

Test 5 Atmospheric air-water test to show the effect of a separator tube

A separator tube was added to the HJTC design to form a collapsed liquid level so that the HJTC sensor directly measures liquid inventory under all simulated two phase conditions. In October, 1980, atmospheric air-water tests were performed with HJTC sensor and the separator tube. The results demonstrated that the separator tube did form a collapsed liquid level and the HJTC output did accurately indicate liquid inventory. This test verified that the HJTC instrument, which includes the HJTC, the splash shield, and the separator tube, is a viable measuring device for liquid inventory.

The Phase 2 test program will consist of high pressure and temperature tests on the HJTC instrument. These tests will provide input for the C-E HJTC instrument design and manufacturing effort. The Phase 2 test program is expected to be completed in early 1981.

The Phase 3 test program will consist of high temperature and pressure testing of the manufactured prototype system HJTC probe assembly and processing electronics. Verification of the HJTC system prototype will be the goal of this test program. The Phase 3 test program is expected to be completed by the end of 1981.

4.3 CORE EXIT THERMOCOUPLES

No verification testing of the CETs is planned. A study at ORNL was performed to test the response of CETs under simulated accident conditions (Reference 6). This test showed that the instruments remained functional up to 2300°F. This test along with previous reactor operating experience verify the response of CETs.

4.4 PROCESSING AND DISPLAYS

The final processing and display design for the ICC detection system has not been completed. As the design effort proceeds, design evaluations will be performed prior to installation.

The integrated functional design of the processing and displays is to be completed by June 1981.

5.0 SYSTEM QUALIFICATION

The qualification program for the ICC Detection System instrumentation has not been completely defined. However, plans are being developed based on the following three categories of ICC instrumentation:

1. Sensor instrumentation within the pressure vessel.
2. Instrumentation components and systems which extend from the primary pressure boundary up to and including the primary display isolator and including the backup displays.
3. Instrumentation systems which comprise the primary display equipment.

A preliminary outline of a qualification program for each classification is given below.

The in-vessel sensors will meet the NUREG-0737, Appendix A guide to install the best equipment available consistent with qualification and schedular requirements. Design of the equipment will be consistent with the guidelines of Appendix A as well as the clarification and Attachment 1 to Item II.F.2 in NUREG-0737. Specifically, instrumentation will be designed such that they meet appropriate stress criteria when subjected to normal and design basis accident loadings. Verification testing will be conducted to confirm operation at DBA (as defined by C-E) pressure and temperature conditions. Seismic testing to safe shutdown conditions will verify function after being subjected to the seismic loadings.

The out-of-vessel instrumentation system, up to and including the primary display isolator, and the backup displays will be environmentally qualified in accordance with IEEE-323-1974 as interpreted by Combustion Engineering Document, CENPD-255, "Qualification of Combustion Engineering Class 1E Instruments." This document describes the method which will be used to qualify out-of-vessel Class 1E equipment.

Plant-specific containment temperature and pressure design profiles will be utilized where appropriate in these tests. This equipment will also be seismically qualified according to IEEE-STD-344-1975. CEN-99(S) ~~CENPD-182~~, "Seismic Qualification of ~~C-E~~ NSSS Supplied Instrumentation Equipment, Combustion Engineering, Inc" (August 1978) describes the methods used to meet the criteria of this document.

The primary display will not be designed as a Class 1E system, but will be designed for high reliability; thus it will not be qualified environmentally or seismically to Class 1E requirements nor will it meet the single failure criteria of Appendix A, Item 2. Post-accident maintenance accessibility will be included in the design. The quality assurance provisions of Appendix A, Item 5 do not apply to the primary display according to NUREG-0737. However, the computer driven primary display system will be separated from the Class 1E sensors, processing and backup display equipment by means of an isolation device which will be qualified to Class 1E criteria.

6.0 OPERATING INSTRUCTIONS

Guidelines for reactor operators to use to detect ICC and take corrective action has been developed by the C-E Owners Group and submitted to NRC for review (Ref. 8). These guidelines have been used to review and revise the plant emergency procedures for San Onofre Units 2 and 3. In addition, the C-E Owners Group has developed reactor operator training materials concerning ICC. The training staff attended a training seminar conducted by C-E in November, 1979, to initiate the San Onofre ICC training program.

The C-E Owners Group is defining a program for development of further emergency procedure guidelines and operator training materials associated with the ICC Detection System described in Section 3. This program is expected to provide these guidelines and training materials during 1981. A more specific schedule is subject to finalization of the ICC Detection System design, specifically the instrument displays.

REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements," U. S. Nuclear Regulatory Commission, November, 1980.

2. CEN-117, "Inadequate Core Cooling - A Response to NRC I E Bulletin 79-06C, Item 5 for Combustion Engineering Nuclear Steam Supply Systems," Combustion Engineering, October, 1979.

3. CEN-125, "Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems," Combustion Engineering, December, 1979.

4. Anderson, R. L., Banda, L. A., Cain, D. G., "Incore Thermocouple Performance Under Simulated Accident Conditions", presented at IEEE Symposium, November, 1980.

5. Letter C-E Owners Group to NRC, "C-E Generic Emergency Procedure Guidelines," December 10, 1980.

APPENDIX A

SUBCOOLED MARGIN MONITOR

A two-channel sub-cooled margin monitor system (SMMS) is being incorporated into the San Onofre Units 2&3 design to provide on-line control room indication of reactor coolant saturation conditions. The SMMS is designed for use as post-accident monitoring instrumentation and is designed to safety grade Seismic Class 1, and Quality Class II standards. The SMMS for San Onofre Units 2 & 3 will be operational prior to fuel load.

A. SYSTEM DESCRIPTION

The SMMS is a two-channel on-line microprocessor based system which uses reactor coolant process signals to provide a continuous indication of the margin from saturation conditions. As shown in figure II.F.2-1, a combination of primary coolant cold leg and hot leg RTD temperature sensors provides wide range primary coolant temperature input to the SMM. Temperature is monitored in a cross-core arrangement (i.e., T_H Loop 1, T_c Loop 2A, and T_c Loop 1A into one monitor) to minimize the impact of a single failure. Primary pressure inputs to the SMMS are provided by two wide range safety grade redundant pressurizer pressure channels. The SMMS microprocessor contains steam tables and interpolation routines for calculating saturation temperatures and pressures. The microprocessor compares the calculated saturation values to the temperature and pressure values from the process inputs and calculates a margin to saturation. Either the temperature or pressure margin can be displayed on the main control board mounted digital display meters which are part of the SMM system. Additionally, the system provides a low margin alarm function with a pre-programmed alarm setpoint for San Onofre Units 2&3. The alarm function of the system inputs the plant annunciation system to provide a low margin alarm to the control room operators.

B. DESIGN SPECIFICATIONS

The following specifications are applicable to the subcooled margin monitoring system.

1. Power Requirements: The two channels of microprocessors, indicators, and process equipment are powered from two different safety grade 120V instrument buses which receive their power from the vital bus power supplies.

2. Process Inputs: (See figure II.F.2-1)

Pressurizer Pressure	0 to 3000 lb/in^2a
RCS Cold Leg Temperature	0 to 710F
RCS Hot Leg Temperature	0 to 710F

3. Low Margin Alarm Setpoint:

   30F Subcooled

4. Microprocessor Range:

   0 to 200F Subcooled

APPENDIX B

HEATED JUNCTION THERMOCOUPLE SYSTEM

B. SYSTEM DESCRIPTION

The Heated Junction Thermocouple System (HJTC) that is planned to be installed in San Onofre Units 2&3 consists of two separate channels of instrumentation which meet the design requirements for a post-accident monitoring system. The sensors are internal to the reactor vessel and the system performs indication, recording, and alarm functions. Each channel consists of a sensing probe, signal processing equipment, and control room display and alarm equipment.

The system uses the ~~principle of~~ heated junction thermocouples (HJTC) to detect a steam-water interface in the reactor vessel and thereby determines the coolant level. Specifically, each sensor consists of two thermocouples connected in electrical opposition, one with a heater in close proximity and one without. When the heated thermocouple is surrounded by liquid, its output is approximately the same as the unheated thermocouple in the same liquid. However, if the heated thermocouple is surrounded by steam or voids, the heating effect of the heater drastically changes the thermocouple output and the presence of a steam-liquid interface can be detected. Each sensor probe assembly contains at least four HJTC sensors arranged in a vertically oriented array and mounted in the upper guide structure region of the reactor vessel. This vertical sensor arrangement thereby enables the system to accurately monitor coolant level in the upper guide structure region.

C. TECHNICAL DESCRIPTION OF THE REACTOR VESSEL INTERNALS CHANGE

The changes concern hardware modifications internal to the reactor vessel which will serve as a holder and guide path for level detector assemblies. The design of the holders will facilitate future use of the level detectors.

Basically, three major components are affected by the modification. These include the upper guide structure assembly, the instrument support plate assembly, and the in-core instrumentation nozzle. The upper guide structure changes include two instrument guide tubes support brackets and lead-in funnels as shown on figure II.F.2-2. The instrument support plate is being modified to accommodate the thimble cluster assembly as shown in figure II.F.2-3. Additional penetrations are being added in each of the two ICI nozzle flanges.

When the above changes are complete, San Onofre Unit 2 will have provisions for two level detector assemblies located as shown in figure II.F.2-4. It is expected that an identical arrangement will be added to Unit 3.

FSAR sections that are affected by this change are sections 4.1 and 4.2, and figure 4.6-1.

D. IMPLEMENTATION SCHEDULE

The San Onofre Units 2&3 RVLM system will be fully implemented ~~by January 1, 1982~~ during the first refueling outage.

REFERENCE

FSAR sections 4.1 and 4.2 and FSAR figure 4.6-1.

Note: No changes to Figures II.F.2-1, 2-2, 2-3, 2-4

APPENDIX - C

Core Exit Thermocouple System

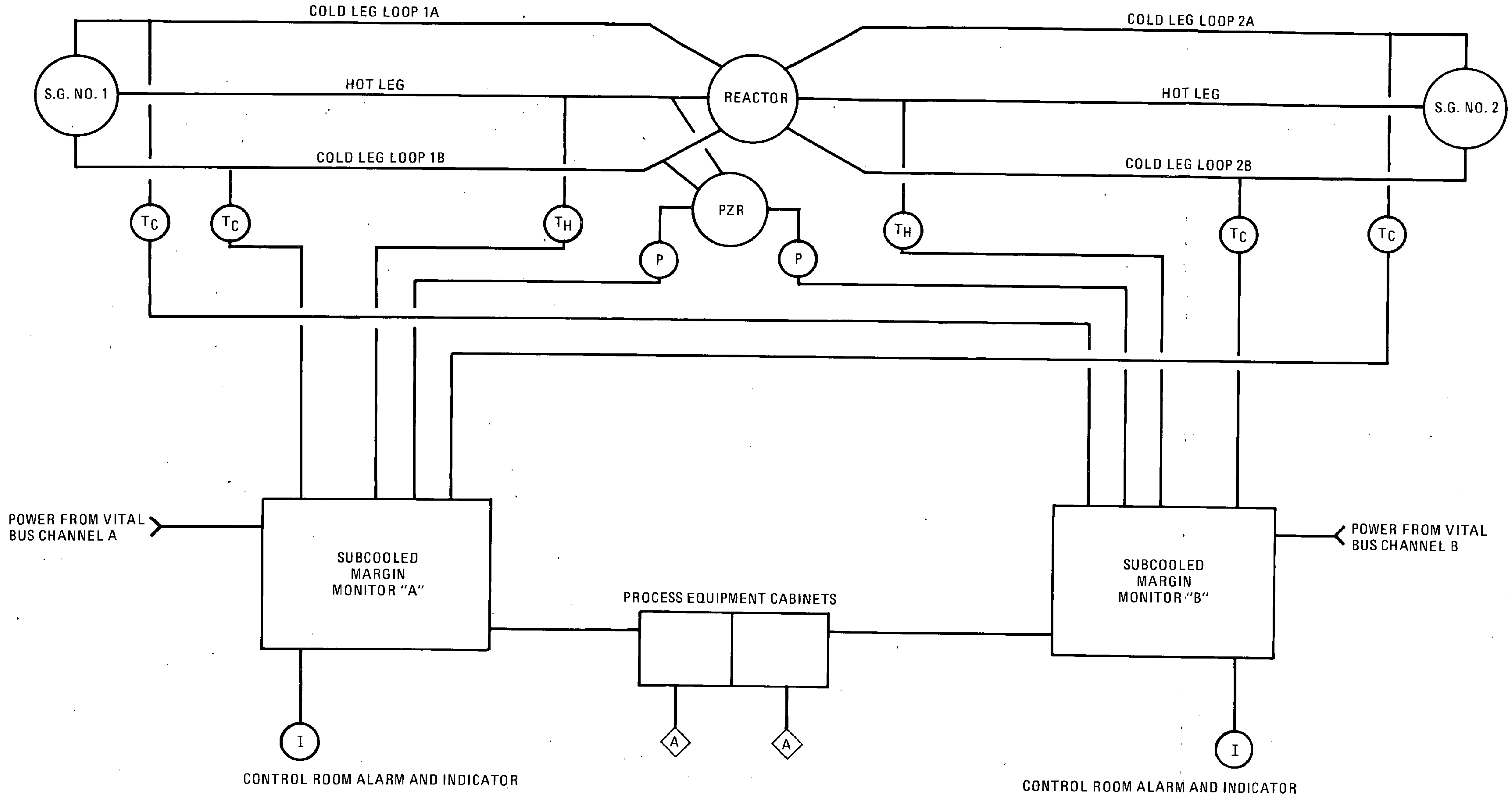
(information to be provided
by February 16, 1981)

II.F.2.1-43

APPENDIX - C

Core Exit Thermocouple System

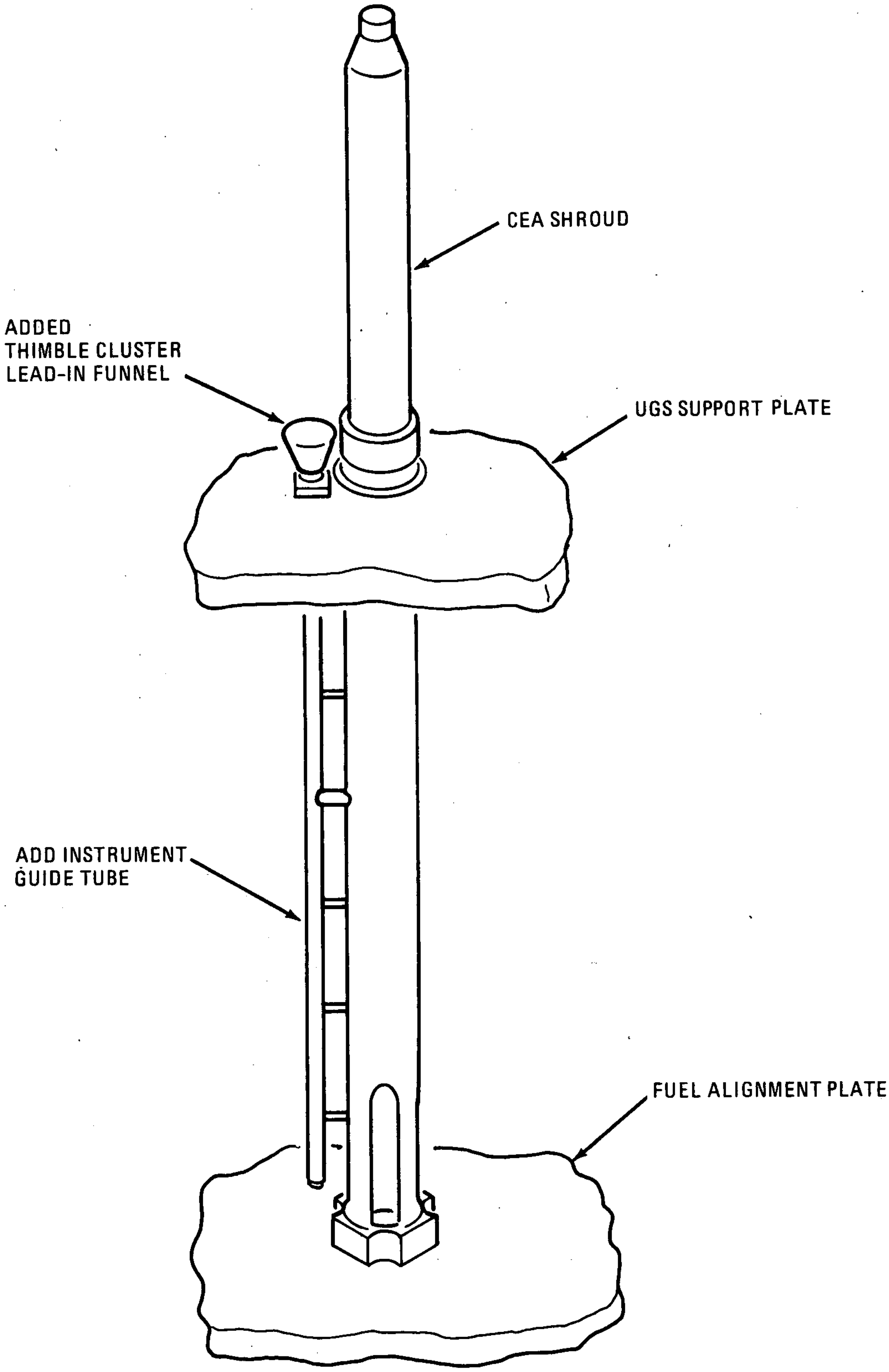
(information to be provided
by February 16, 1981)



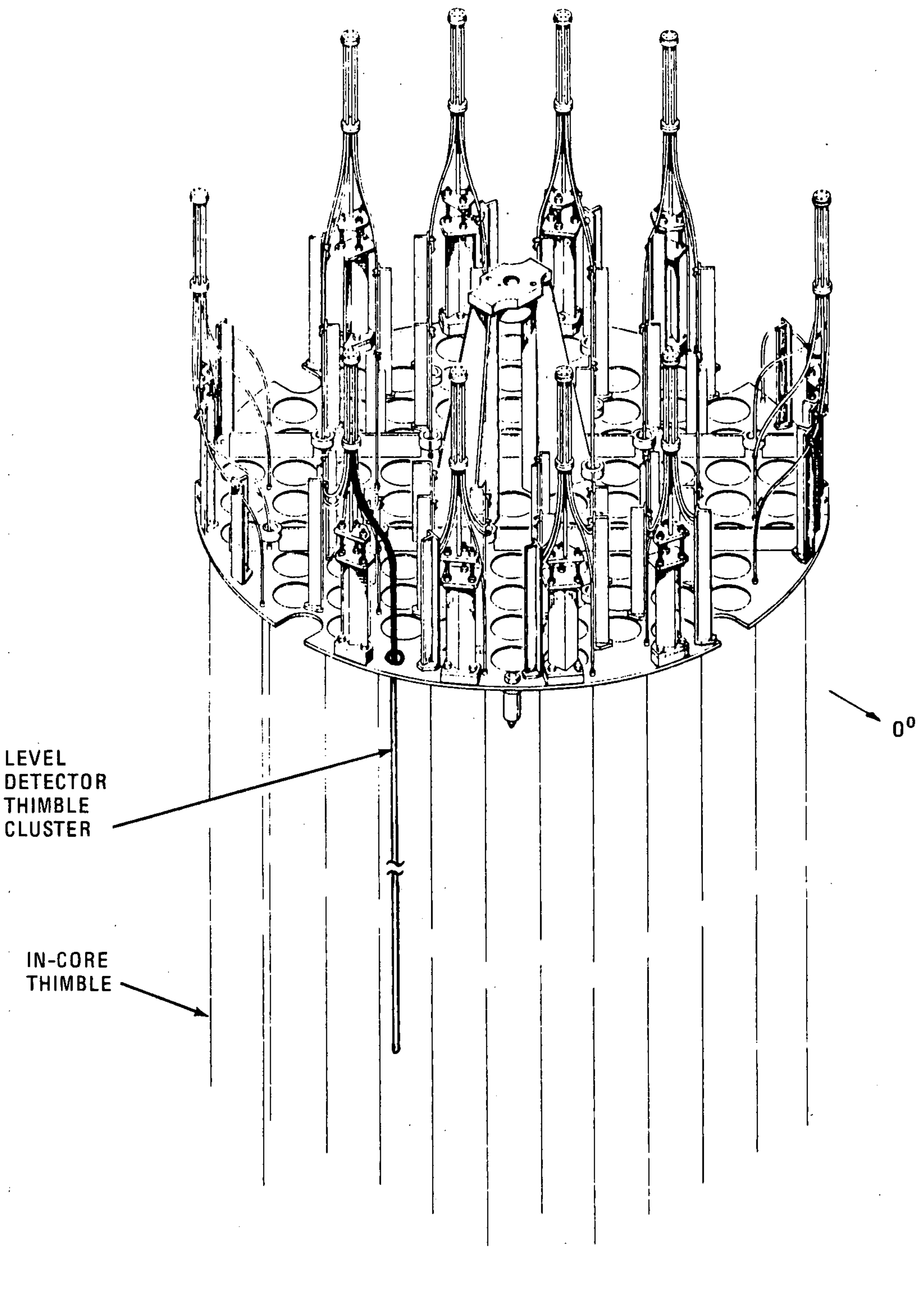
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

SMM SYSTEM LOGIC DIAGRAM

Figure II.F.2-1



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
REACTOR VESSEL LEVEL MONITORING SYSTEM INSTRUMENT GUIDE TUBE
Figure II.F.2-2



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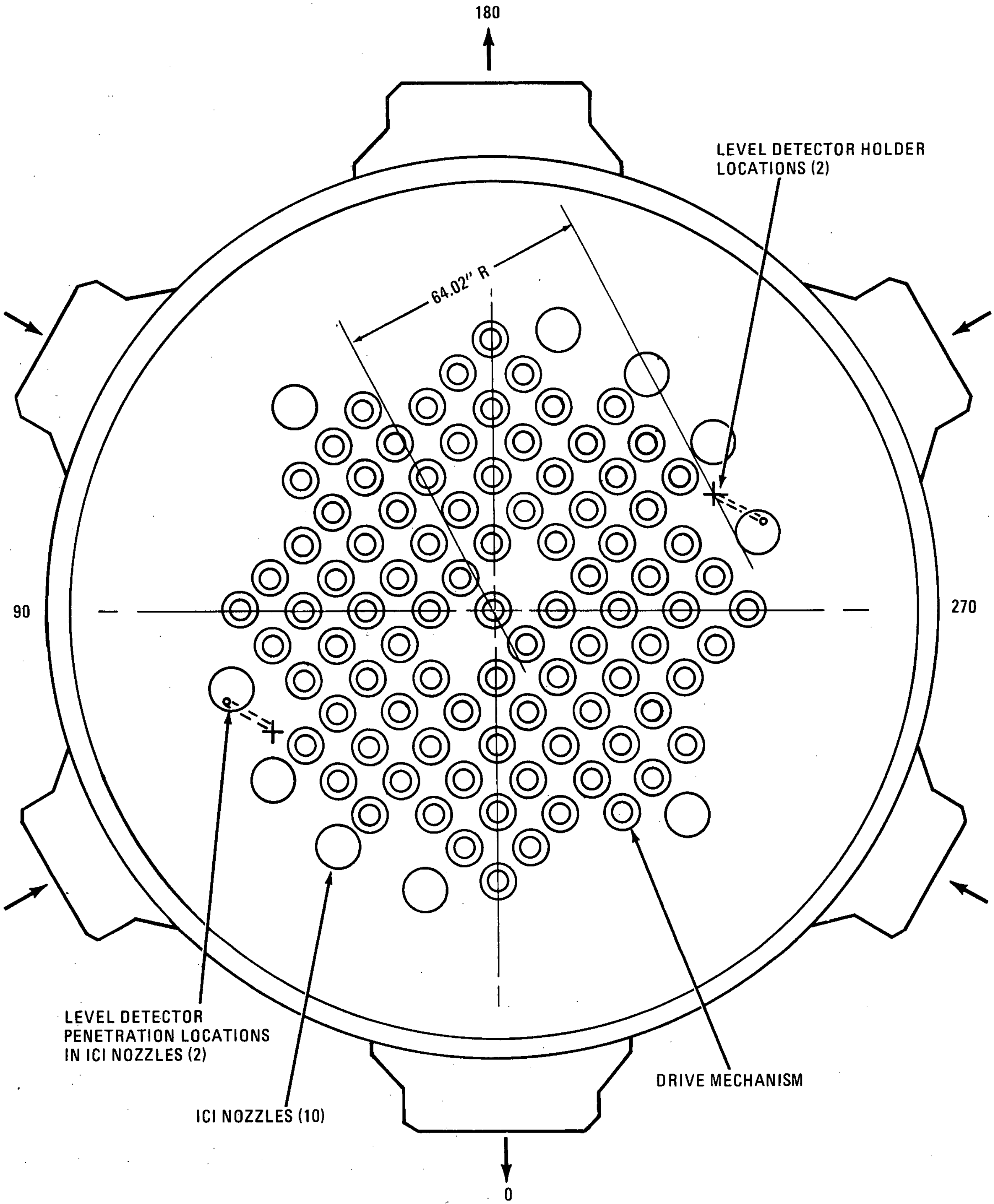
Units 2 & 3

INSTRUMENT SUPPORT PLATE

Figure II.F.2-3



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Units 2 & 3

REACTOR VESSEL HEAD

Figure II.F.2-4

II.G.1 - NUREG 0737 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

REQUIREMENT

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grace requirements. 23

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Clarification

1. Although the primary concern resulting from lessons learned from the accident at TMI is that the PORV block valves must be closable, the design should retain, to the extent practical, the capability to also open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from the source supplying the PORV.

3. Any changeover of the PORV and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs in which instrument air is needed for operation, the electrical power supply should be required to have the capability to be manually connected to the emergency power sources.

RESPONSE

1. NRC Positions 1, 2, and 3

   San Onofre Units 2 and 3 do not utilize power-operated relief valves or block valves; accordingly these items are not applicable.

2. NRC Position 4

   The current station design complies with the stated position requirements. Pressurizer level indication is provided by two redundant, Class IE powered instrument channels. The buses have the capability of being energized from either offsite power or the emergency diesel generators. See FSAR section 7.5 for a discussion of safety-related display instrumentation.

Documentation in response to the requirements of II.G.1 are as follows:

23 System Design Description - FSAR Section 7.5; Tables 7.5-1; 7.7.1.2.2, Figure 5.1-9

Logic Diagrams - FSAR Table 1.7-1

Electrical Schematics - FSAR Table 1.7-1

Test Procedures will be developed as part of the normal startup test program and will be available for NRC staff review.

Technical Specifications will include operability requirements for the pressurizer level indication instruments.

REFERENCES

FSAR subsection 5.4.10 Figure 5.1-9, Table 1.7-1, 8.1.3
FSAR Section 7.5, Table 7.5-1, and 7.7.1.2.2

No FSAR changes were made.

II.K.1 - NUREG 0660 IE BULLETINS ON MEASURES TO MITIGATE SMALL-BREAK LOCAs AND LOSS OF FEEDWATER ACCIDENTS

REQUIREMENT

NRR will require all operating license applicants to evaluate their plants against the requirements specified in applicable IE Bulletins and not otherwise addressed in this Action Plan, and to take corrective actions as necessary prior to fuel loading. Ultimately, these requirements will be modified by NRR and SD, as appropriate, and required of all plants as preconditions for receipt of an operating license.

NUREG 0694, TMI-Related Requirements for New Operating Licenses, lists the following two IE Bulletin requirements (item numbers from Table C.1 of NUREG 0660) applicable to Combustion Engineering plants:

C.1.5. Review all valve positions, positioning requirements, positive controls, and related test and maintenance procedures to assure proper ESF functioning.

C.1.10 Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known.

RESPONSE

A review of all safety-related valve positions, positioning requirements, and positive controls to ensure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features will be completed and documented by October 1980. Appropriate related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic surveillance, will be revised as necessary.

A review of all procedures concerning the removal from and return to service of safety-related systems to assure that their operability status is known by the operators will be completed and documented by October 1980. Procedures will be modified as necessary to ensure that:

- Redundant systems are verified operable prior to removing a safety-related system from service.
- Safety-related systems are verified operable when returned to service.
- Operators are explicitly notified of any actions changing the status of a safety-related system.

REFERENCES

None.

II.K.2.13 - NUREG 0737 THERMAL MECHANICAL REPORT--EFFECT OF HIGH-PRESSURE INJECTION VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

REQUIREMENT

Position

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

Clarification

The position deals with the potential for thermal shock of reactor vessels resulting from cold safety injection flow. One aspect that bears heavily on the effects of safety injection flow, is the mixing of safety injection water with reactor coolant in the reactor vessel. B&W provided a report on July 30, 1980 that discussed the mixing question and the basis for a conservative analysis of the potential for thermal shock to the reactor vessel. Other PWR vendors are also required to address this issue with regard to recovery from small breaks with an extended loss of all feedwater. In particular, demonstration shall be provided that sufficient mixing would occur of the cold high-pressure injection (HPI) water with reactor coolant so that significant thermal shock effects to the vessel are precluded. 23

RESPONSE

Southern California Edison (SCE) is participating in a C-E Owners Group evaluation of the generic applicability of these requirements. If sufficient generic bases are determined to exist, the C-E Owners Group will consider preparation of a generic response by January 1, 1982. SCE will participate in this C-E Owners Group activity and provide a response to these requirements by January 1, 1982.

II.K.2.17 - NUREG 0737 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

REQUIREMENT

Position

Analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients.

Clarification

The background for this concern and a request for this analysis was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from R. W. Reid, NRC, to all P&W operating plants, dated January 9, 1980. 23

The results of this evaluation have been submitted by the B&W licensees and is presently undergoing staff review.

RESPONSE

Southern California Edison (SCE) is participating in a C-E Owners Group evaluation of the generic applicability of these requirements. If sufficient generic bases are determined to exist, the C-E Owners Group will consider preparation of a generic response by January 1, 1982. SCE will participate in this C-E Owners Group activity and provide a response to these requirements by January 1, 1982.

II.K.2.19 - NUREG 0737 SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS

REQUIREMENT

Position

Provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of main feedwater.

Clarification

This requirement was originally sent to the Babcock and Wilcox (B&W) licensees in a letter from D. F. Ross, Jr., NRC, to all B&W operating plants, dated August 21, 1979. 23

The results of this analysis has been submitted by the B&W licensees and is presently undergoing staff review.

RESPONSE

Southern California Edison (SCE) is participating in a C-E Owners Group evaluation of the generic applicability of these requirements. If sufficient generic bases are determined to exist, the C-E Owners Group will consider preparation of a generic response by January 1, 1982. SCE will participate in this C-E Owners Group activity and provide a response to these requirements by January 1, 1982.

II.K.3.1 - NUREG 0737 INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

REQUIREMENT

Position

All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to assure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation. 23

Clarification

Implementation of this action item was modified in the May 1980 version of NUREG-0660. If such studies confirm that the subject system is necessary, the change delays implementation of this action item until after the studies specified in TMI Action Plan Item II.K.3.2 have been completed.

RESPONSE

The above requirements are not applicable to San Onofre Units 2 and 3 as PORV's are not part of the plant design.

II.K.3.2 - NUREG 0737 **REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM**

REQUIREMENT

Position

1. The licensee should submit a report, for staff review, documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

2. Safety valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

Clarification

Based on its review of feedwater transients and small LOCAs for operating plants, the Bulletins and Orders Task Force in the Office of Nuclear Reactor Regulation recommended that a report be prepared and submitted for staff review which documents the various actions that have been taken to reduce 23
the probability of a small-break LOCA caused by a stuck open PORV and show how these actions constitute sufficient improvements in reactor safety. Action Item II.K.3.2 of NUREG-0660, published in May 1980, changed the implementation of this recommendation as follows: In addition to modifications already implemented on PORVs, the report specified above should include safety examination of an automatic PORV isolation system identified in Action Item II.K.3.1.

Modifications to reduce the likelihood of a stuck open PORV will be considered sufficient improvements in reactor safety if they reduce the probability of a small-break LOCA caused by a stuck open PORV such that it is not a significant contributor to the probability of a small-break LOCA due to all causes. (According to WASH-1400, the median probability of a small-break LOCA S_2 with a break diameter between 0.5 in. and 2.0 in. is 10-3 per reactor year with a variation ranging from 10-2 to 10-4 per reactor year.)

The above specified report should also include an analysis of safety valve failures based on the operating experience of the pressurized-water-reactor (PWR) vendor designs. The licensee has the option of preparing and submitting either a plant-specific or a generic report. If a generic report is submitted, each licensee should document the applicability of the generic report to his own plant.

Based on the above guidance and clarification, each licensee should perform an analysis of the probability of a small-break LOCA caused by a stuck open PORV or safety valve. This analysis should consider modifications which have been made since the TMI-2 accident to improve the probability. This analysis shall evaluate the effect of an automatic PORV isolation system specified in Action Plan Item II.K.3.1. In evaluating the automatic PORV isolation system, the potential of causing a subsequent stuck open safety valve and the overall effect on safety (e.g., effect on other accidents) should be examined.

Actual operational data may be used in this analysis where appropriate. The basis for any assumptions used should be clearly stated and justified.

The results of the probability analysis should then be used to determine whether the modifications already implemented have reduced the probability of a small-break LOCA due to a stuck open PORV or safety valve a sufficient
23 amount to satisfy the criterion stated above, or whether the automatic PORV isolation system specified in Action Item II.K.3.1 is necessary.

In addition to the analysis described above, the licensee should compile operational data regarding pressurizer safety valves for PWR vendor designs. These data should then be used to determine safety valve failure rates.

The analyses should be documented in a report. If this requirement is implemented on a generic basis, each licensee should review the appropriate generic report and document its applicability to his own plant(s). The report and the documentation of applicability (where appropriate) should be submitted for NRC staff review by the specified date.

RESPONSE

The above requirements are not applicable to San Onofre Units 2 and 3 as PORV's are not part of the plant design.

II.K.3.3 - NUREG 0737 REPORTING SV & RV FAILURES AND CHALLENGES

REQUIREMENT

Position

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report. 23

RESPONSE

Failures and challenges to safety and relief valves will be reported as required by the reporting requirements of the San Onofre Units 2&3 technical specifications.

REFERENCES

None

II.K.3.5 - NUREG 0737 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT

REQUIREMENT

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Clarification

This action item has been revised in the May 1980 version of NUREG-0660 to provide for continued study of criteria for early reactor coolant pump trip. Implementation, if any is required, will be delayed accordingly. As part of the continued study, all holders of approved emergency core cooling (ECC) models have been required to analyze the forthcoming LOFT test (L3-6). 23
The capability of the industry models to correctly predict the experimental behavior of this test will have a strong input on the staff's determination of when and how the reactor coolant pumps should be tripped.

RESPONSE

San Onofre Units 2 and 3, through participation in the C-E Owners' Group, is continuing to study the effect of RCPs on small break LOCA and the possible need for an automatic RCP trip. A report titled "Response to NRC Bulletin 79-06C Items 2 and 3 for Combustion Engineering Nuclear Steam Supply Systems" (CEN-115) has been provided to the NRC by the C-E Owners' Group. CEN-115 contains the results of a generic study of the influence of RCPs on small break transients. The major conclusions of this study are:

1. Continued operation of the Reactor Coolant Pumps (RCPs) after a small break LOCA in the hot leg of the Reactor Coolant System results in more and longer uncoverage of the reactor core by two-phase liquid than when the RCPs are turned off after the break.

2. For non-LOCA events, tripping of the RCPs in general has the undesirable effect of reducing margins to fuel failure and of increasing radiological releases. However, the review indicates that consequences of non-LOCA events will be acceptable with RCP trip following reactor coolant system depressurization to the SIAS setpoint, provided the RCPs are tripped at least five seconds after the control rods are fully inserted.

3. The calculational models used in the analyses described in this report appear reasonable and are qualitatively supported by Semiscale test results. However, they have been in use only for a very short time compared to the length of time the models of the evaluation model have been in use. Thus, while there is a high degree of confidence in the general conclusions of this report, the detailed quantitative results of specific analyses may change as further model refinements are developed.

Recognizing the need for increased credibility of the analytical models used to predict the effects of powered RCPs, San Onofre Units 2 and 3 has commissioned C-E through the C-E Owners Group to perform model development and verification. Included in this effort is an analysis of the LOFT test L3-6. Following model verification through comparison to integral test
23 data, the need for an automatic RCP trip will be reassessed.

SCE has revised the appropriate emergency procedures to reflect the requirement to manually trip the RCP's following a depressurization to the SIAS set point (Referenced letter). SCE has chartered C-E to perform an analysis to determine what RCS pressure below the SIAS setpoint the RCP's can be tripped and still provide results which meet NRC criteria.

REFERENCE

SCE letter dated 11/12/80 (K. Baskin to F. Miraglia (NRC) transmitting four Emergency Operating Instructions.

II.K.3.17 - NUREG 0737 REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

REQUIREMENT

Position

Several components of the emergency core cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Clarification

The present technical specifications contain limits on allowable outage times for ECC systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification requirements but have a high unavailability because of frequent outages within the allowable technical specifications. 23

The licensees should submit a report detailing outage dates and length of outages for all ECC systems for the last 5 years of operation, including causes of the outages. This report will provide the staff with a quantification of historical unreliability due to test and maintenance outages, which will be used to determine if a need exists for cumulative outage requirements in the technical specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC systems or components involved in the outage; and (4) corrective action taken. Test and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

Applicant for an operating license shall establish a plan to meet these requirements.

RESPONSE

SCE will establish a plan for reporting ECCS cumulative outage times. This plan will be established prior to fuel load.

23 REFERENCE

None.

II.K.3.25 - NUREG 0737 - EFFECT OF LOSS OF ALTERNATING-CURRENT POWER ON PUMP SEALS

REQUIREMENT

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Clarification

The intent of this position is to prevent excessive loss of reactor coolant system (RCS) inventory following an anticipated operational occurrence. Loss of ac power for this case is construed to be loss of offsite power. If seal failure is the consequence of loss of cooling water to the reactor coolant pump (RCP) seal coolers for 2 hours, due to loss of offsite power, one acceptable solution would be to supply emergency power to the component cooling water pump. This topic is addressed for Babcock and Wilcox (B&W) reactors in Section II.K.2.16. 23

RESPONSE

San Onofre Units 2 and 3 Reactor Coolant Pumps have been operationally tested to demonstrate satisfactory seal performance with seal cooling water shut off for 30 minutes with the pump operating. Results of this test are documented in the response to NRC Questions 010.48 and 212.159. A test to specifically show that the San Onofre Units 2 and 3 RCP seal assemblies would not lose function following a loss of seal cooling when the pumps are not in operation for 2 hours has not been performed. Based on the 30 minute operational test it was demonstrated that the seals would not lose function (i.e., gross leakage) but the seal assemblies did require refurbishment following the test. It is the judgement of Combustion Engineering that the RCP seals would not lose function following a loss of power two hours in duration.

II.K.3.30 - NUREG 0737 REVISED SMALL BREAK LOSS-OF-COOLANT-ACCIDENT METHODS TO SHOW COMPLIANCE WITH 10 CFR PART 50, APPENDIX K

REQUIREMENT

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Clarification

As a result of the accident at TMI-2, the Bulletins and Orders Task Force was formed within the Office of Nuclear Reactor Regulation. This task force was charged, in part, to review the analytical predictions of feed-water transients and small-break LOCAs for the purpose of assuring the continued safe operation of all operating reactors, including a determination of acceptability of emergency guidelines for operators. 23

As a result of the task force reviews, a number of concerns were identified regarding the adequacy of certain features of small-break LOCA models, particularly the need to confirm specific model features (e.g., condensation heat transfer rates) against applicable experimental data. These concerns, as they applied to each light-water reactor (LWR) vendor's models, were documented in the task force reports for each LWR vendor. In addition to the modeling concerns identified, the task force also concluded that, in light of the TMI-2 accident, additional systems verification of the small break LOCA model as required by II.4 of Appendix K to 10 CFR 50 was needed. This included providing predictions of Semiscale Test S-07-10B, LOFT Test (L3-1), and providingg experimental verification of the various modes of single-phase and two-phase natural circulation predicted to occur in each vendor's reactor during small break LOCAs.

Based on the cumulative staff requirements for additional small break LOCA model verification, including both integral system and separate effects verification, the staff considered model revision as the appropriate method for reflecting any potential upgrading of the analysis methods.

The purpose of the verification was to provide the necessary assurance that the small break LOCA models were acceptable to calculate the behavior and consequences of small primary system breaks. The staff believes that this assurance can alternatively be provided, as appropriate, by additional justification of the acceptability of present small break LOCA models with regard to specific staff concerns and recent test data. Such justification could supplement or supersede the need for model revision.*

The specific staff concerns regarding small break LOCA models are provided in the analysis sections of the B&O Task Force reports for each LWR vendor, (NUREG-0635, -0555, -0626, -0611, and -0623). These concerns should be reviewed in total by each holder of an approved emergency core cooling system (ECCS) model and addressed in the evaluation as appropriate.

The recent tests include the entire Semiscale small break test series and LOFT Tests (L3-1) and (L3-2). The staff believes that the present small break LOCA models can be both qualitatively and quantitatively assessed against these tests. Other separate effects tests (e.g., ORNL core uncovery tests) and future tests, as appropriate, should also be factored into this assessment.

Based on the preceding information, a detailed outline of the proposed program to address this issue should be submitted. In particular, this submittal should identify (1) which areas of the models, if any, the licensee intends to upgrade, (2) which areas the licensee intends to address by further justification of acceptability, (3) test data to be used as part of
23 the overall verification/upgrade effort, and (4) the estimated schedule for performing the necessary work and submitting this information for staff review and approval.

RESPONSE

Southern California Edison (SCE) is participating in the C-E Owners Group to sponsor preparation of prediction of LOFT Test L3-6. Documentation of the C-E Owners Group's plan for such test prediction and response to related NRC requirements is provided in the referenced letter.

Because of the importance of these tests in evaluation of the adequacy of present small break LOCA models, SCE feels that it is inappropriate to consider further model documentation or changes until completion of the NRC review of these test predictions. Therefore, SCE will continue to participate as a member of the C-E Owners Group to support preparation of the predictions of the LOFT Test L3-6.

REFERENCE

G. Liebler letter to P. S. Check dated July 31, 1980 concerning "Test Analysis of LOFT Small Break Test L3-6".

II.K.3.31 - NUREG 0737 PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR PART 50.46

REQUIREMENT

Position

Plant specific calculations using NRC approved models for small break loss-of-coolant accidents (LOCAs) as described in item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees. 23

Clarification

See "Clarification" for item II.K.3.30.

RESPONSE

The San Onofre Units 2 and 3 small break LOCA analysis has been completed and results are documented in FSAR Sections 6.3.3.3 and 15.6. SCE will perform a supplemental plant specific analysis if the results of LOFT Test L3-6 (discussed in item II.K.3.30 response) or other information indicate the need for an additional analysis.

III.A.1.2 - NUREG 0660 UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

REQUIREMENT

A. ONSITE TECHNICAL SUPPORT CENTER (NUREG-0578, ITEM 2.2.2.b)

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center. Records that pertain to the as-built conditions and layout of structures, systems and components shall be readily available to personnel in the TSC.

Clarification (NRC Letter Dated November 9, 1979)

1. By January 1, 1980, each licensee should meet items A-G that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981. 23

   a. Establish a TSC and provide a complete description.

   b. Provide plans and procedures for engineering/management support and staffing of the TSC.

   c. Install dedicated communications between the TSC and the control room, near site emergency operations center, and the NRC.

   d. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets or evacuation to the control room).

   e. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters, necessary for assessment in the TSC.

   f. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable.

g. Submit to the NRC a longer range plan for upgrading the TSC to meet all requirements.

2. Location

It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information during an emergency. The center should be located onsite, i.e., within the plant security boundary. The greater the distance from the CR, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. Physical Size and Staffing

The TSC should be large enough to house 25 persons, necessary engineering data and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

4. Activation

The center should be activated in accordance with the "Alert" level as
23 defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September, 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = υ defined as either reactor or turbine trip). The Shift Technical Advisor should be consulted on the "Notification of Unusual Event" however, the activation of the TSC is discretionary for that class of event.

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirements but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the capability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to assure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To insure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be given to assure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

7. <u>Technical Data</u>

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum, data (historical in addition to current status) should be available to permit the assessment of:

a. Plant Safety Systems Parameters for:

   1. Reactor Coolant System

   2. Secondary System (PWRs)

   3. ECCS Systems 23

   4. Feedwater and Makeup Systems

   5. Containment

b. In-Plant Radiological Parameters for:

   1. Reactor Coolant System

   2. Containment

   3. Effluent Treatment

   4. Release Paths

c. Offsite Radiological

   1. Meteorology

   2. Offsite Radiation Levels

8. <u>Data Transmission</u>

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display.

Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the Emergency Operations Center, the NRC, the reactor vendor, etc.

9. <u>Structural Integrity</u>

   a. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

   b. Since the center need not be designed to the same stringent requirements as the Control Room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. <u>Habitability</u>

   The licensee should provide protection for the technical support center personnel from radiological hazards including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4.

23 a. Licensee should assure that personnel inside the technical support center (TSC) will not receive doses in excess of those specified in GDC 19 and SRP 6.4 (i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

   b. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

   c. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be followed except that the systems do not have to be redundant, seismic, instrumented in the control room or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52 and the HEPA's do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

d. Dose reduction measures such as breathing apparatus and potassium iodide tablets can not be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available.

RESPONSE

A. ONSITE TECHNICAL SUPPORT CENTER

In response to the NRC's position to provide upgraded emergency support facilities, San Onofre Units 2&3 will establish an onsite Technical Support Center (TSC). The objective of the TSC is to display and transmit plant status to appropriate individuals in the event of an accident. The applicants will be guided by the staff's positions delineated in NUREG-0696 relative to the emergency response facilities described below. A description of the TSC and its procedures for implementation is given in the following paragraphs.

The center will be habitable to the same degree as the control room for postulated accident condition. The emergency plan will be revised as necessary to incorporate the role and location of the TSC. Records that pertain to the as-built condition and layout of structures, system and components will be readily available to personnel in the TSC as further shown in the following paragraph. 23

1. Response to the Clarification

a. A description of the TSC and its procedures for implementation is given in the following paragraphs.

b. Plans for engineering/management support and staffing of the TSC are addressed in the San Onofre Units 2 and 3 Emergency Plan.

c. Dedicated communication will be installed between the TSC and the control room, near site emergency operations center, and the NRC. Telephone communications are provided sufficient for onsite and offsite emergency use. These include provisions for two NRC lines (to be arranged by NRC), 5 multi-key telephone units with 2 PAX lines and 2 Pacific Telephone lines on each instrument, 1 extension of the existing dedicated automatic ring-down line to the U.S. Marine Corps Fire Station, 1 extension of the existing dedicated automatic ring-down line to the San Clemente City Hall, and a dedicated intercom between the TSC and the Units 2 and 3 control room.

d. Monitors will be provided for both direct radiation and airborne radioactive contaminants. See also paragraph 5, where a Health Physics Computer System is mentioned.

e. Access to Technical Data is provided to the TSC to conform to paragraph 7.

f. Should the TSC become uninhabitable, the function of accident assessment can be performed from the control room.

g. SCE is reviewing the guidance outlined in the Final Report, NUREG-0696 which was submitted to the NRC Commissioners on January 12, 1981. This report will form the basis for SCE's long range plans for upgrading the TSC.

2. Location

The San Onofre Units 2&3 TSC will be located in the visitor's viewing gallery and adjacent instrument repair laboratory on elevation 39'-0" in the control building. This TSC will service both Units 2 and 3, and overlooks the control rooms. Visual contact can be achieved through a glass window. Key TSC personnel may have ready access to the control room if required by descending one flight of stairs (refer to figure III.A.1.2-1).

3. Physical Size and Staffing

The TSC area is sufficiently large enough to accommodate up to 25 persons in addition to the necessary engineering data and displays.
23 Procedures will be developed to delineate the specific disciplines which are required to report to the TSC based on the severity of the emergency.

4. Activation

The TSC will be activated in accordance with the "alert" level as defined in NUREG-0610. The instrumentation displaying the plant status (described below) is continuously operating and thus will provide plant parameter display capability from accident initiation. Data storage is accomplished through use of a 67 megabyte moving head disc, and printing capability for permanent recording is via a 600 LMP line printer.

5. Instrumentation

The plant parameters selected for display in the TSC are accomplished by the critical functions monitoring systems (CFMS) supplied by Combustion Engineering, Inc. This system receives data inputs from the plant, using the existing process instrument loops and certain new wide range monitoring instruments added to meet NUREG-0694 requirements. The software for the CFMS provides interactive color CRT displays of the critical safety parameters and one-line models of key systems that are either required or desirable in the event-recovery scenario. An identical CRT display is provided in the control room. The CFMS also has an alarm/diagnosis heirarchy such that any monitored critical parameter that exceeds a predetermined setpoint will activate an alarm indication on the CRT display along with a code to rapidly identify the specific malfunctioning equipment. These data shall be recorded such that the plant steady-state operating conditions prior to the accident, the transient conditions producing the initiating event, and the plant systems dynamic behavior throughout the course of the accident will be available in the TSC.

Additional parameters that may be required in the TSC, such as in-plant radiation monitors, effluent radiation monitors, and meteorological data are planned to be supplied by CRT readout transmitted from a separate health physics/radiological computer system implemented to meet the emergency operations facility (EOF) and TSC requirements. These data will supplement the CFMS displays. A more detailed description of the CFMS is provided at the end of this section, page III.A.1.2-10 and of the Health Physics Computer System.

6. Instrumentation Power Supply

The CFMS is a non-1E system and utilizes either non-1E process inputs or 1E signals that have been appropriately isolated to prevent any potential degradation of the 1E system. However, the CFMS is supplied by an uninterruptable power supply capable of sustaining operation for 23
at least 90 minutes with subsequent manual connection to a diesel back-up 1E power supply for long term TSC operation consistent with criteria in FSAR paragraph 8.1.4.3. The data logging function of the CFMS is normally in continuous operation.

7. Technical Data

The list of data points provided for display in the CFMS is given in table III.A.1.2-1. In addition, certain plant radiological parameters, effluent radiation monitors, meteorological information, and certain offsite radiation monitors will be provided as part of the Health Physics Computer System. The computer system selected to process the required parameters (which include the above) will display them in the TSC via a CRT display.

The technical information provided for the TSC includes vital plant design documents and drawings, as well as plant procedures and manuals. TSC personnel will have ready access to up-to-date records that include the current Plant Technical Specifications, Plant Operating Instructions, Emergency Operating Instructions, Final Safety Analysis Report, and drawings and diagrams showing the current conditions of plant structures and systems. Documentation will include general arrangement drawings, piping and instrumentation drawings, electrical one line and elementary diagrams, radiation zone drawings and selected vendor drawings.

8. Data Transmission

The CFMS will include a modem that enables the plant parameters monitored to be transmitted via data link to an offsite receiver, and will be transmitted to the future EOF (section III A.1.2.C) to be constructed by the Applicants. The EOF will likewise have a similar data link for the radiological information.

9. Structural Integrity

Since the TSC is located within the control room envelope in the control building, the structure is Seismic Category I. However, equipment within the TSC is not designed to function during and following a DBE, although it has been designed to assure that its failure will not impact any SCI equipment. The control room displays will serve as the back-up to the TSC for a severe seismic event.

10. Habitability

Since the TSC is located within the same environmental control envelope as the control room, the emergency safety-related HVAC functions to mitigate abnormal airborne radiation. The dose to the TSC meets GDC 19 criteria for the same source terms as used to analyze the
23 control room (see discussion for item II.b.2). The in-plant radiation monitors will be displayed in the TSC to alert the occupants of potential high radiation areas.

REQUIREMENT

B. ONSITE OPERATIONAL SUPPORT CENTER (NUREG-0578, Item 2.2.2.c)

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

RESPONSE

An onsite operational support center (OOSC) has been established in the lunch room of the administration, warehouse, and shop (AWS) building at San Onofre Units 2&3. The OOSC will serve as a "holding area" outside of the control room for shift support personnel (other than those in the control room) to report and to receive assignments. The OOSC has communication capability with the control room and will also have access to the station documents and records available to the onsite technical support center (see section III.A.1.2A). This facility is being established with due consideration given to the functional criteria stated in NUREG-0696. In addition, the San Onofre Emergency Plan has been revised to reflect existence and function of the center.

REQUIREMENT

C. NEAR-SITE EMERGENCY OPERATION FACILITY (NUREG-0694)

Designate a near-site emergency operations facility with communications with the plant to provide evaluation of radiation releases and coordination of all onsite and offsite activities during an accident.

Provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite technical support center, and direct display of radiological and meteorological parameters.

RESPONSE 23

An interim emergency operations facility (EOF) for Federal, State, and local personnel has been established at the San Clemente City Hall. The EOF will be provided with dedicated two-way automatic ring down circuit to the San Onofre Units 2&3 onsite technical support center (see section III.A.1.2.A of this report) prior to receipt of the operating license. Furthermore, a CRT display of environmental radiation effects generated by the health physics/radiological computer (described in response to section III.A.1.2A) will be located at the EOF.

In addition, an Emergency Support Center (ESC) has been established in the Administration, Warehouse, and Shop (AWS) building conference room for the support personnel from the main offices to gather following "Alert" or more serious events.

These facilities are interim measures pending completion of a well designed near-site Emergency Operations Facility for which the design is under development. This facility is being designed with due consideration given to the functional criteria stated in NUREG-0696.

REFERENCES

None.

CRITICAL FUNCTION MONITORING SYSTEM

1.0 OVERVIEW

The critical function monitoring system (CFMS) is a computer-based alarm and display system. The CFM monitors plant process parameters and displays the parameters in a "human engineered" format to control room operations personnel and to Technical Support Center personnel. The CFMS also calculates and displays the safety state of the plant. This safety state is defined by seven critical functions (safety state vector). The CFMS alarms deviations in the critical functions and provides a structured display and alarm format that provides for enhanced diagnostic capability during normal and non-normal plant operation.

The CFMS is being installed in San Onofre Units 2&3 with system operation scheduled prior to fuel load.

2.0 DESIGN BASES

The CFMS design bases are divided into three areas: functional, hardware, and software.

2.1 FUNCTIONAL DESIGN BASES

23

A. The CFMS shall provide the capability to display the status of the following critical functions:

1. Core Reactivity Control
2. Core Heat Removal Control
3. Reactor Coolant System Inventory Control
4. Reactor Coolant System Pressure Control
5. Reactor Coolant System Heat Removal Control
6. Containment Pressure/Temperature Control
7. Containment Isolation

B. The CFMS shall alarm deviations of the critical functions.

C. The CFMS shall provide the user with concise, understandable, integrated information to assist in assessing plant status during all modes of plant operation. The CFMS displays shall utilize proven human-engineering principles such as displays, alarms, and user input capability.

1. Displays:

The primary user interface to the CFMS is through multi-colored cathode ray tube (CRT) display stations. Each display station is capable of providing any one of the CFMS fixed format displays. Two CRT display stations for each unit will be located in the TSC, and at least one CRT display station in each unit's control room.

2. Display Style

a. The CFMS utilizes the display methodology developed for the Combustion Engineering Nuplex 80TM Advanced Control Center, including:

(1) Symbology

(2) Alarm Color Code

(3) Operational Color Code

(4) Page Formats

(5) Page Number Scheme 23

(6) Numeric Formats

(7) Dynamic Behavior

b. The CFMS shall utilize loop mimic displays. When loop mimics are not possible, a left to right, top to bottom flow shall be assumed.

3. Hierarchy

a. In order to effectively organize the information presented by the CFMS a top down, three level hierarchy is used.

b. The CFMS display pages are arranged in a three level hierarchy which consists of:

(1) Level I - Overview Information

(2) Level II - Systems Information

(3) Level III - Subsystem and Component Information (see figure III.A.1.2-2)

c. Level 1 display pages provide overview information about both the plant and the CFMS. Level 1 displays are primarily alphanumeric.

d. The Level I display pages include:

(1) Display Directory

An alphanumeric display which lists the display page titles and page numbers.

(2) Current Alarm List

An alphanumeric display which lists parameter alarms in chronological order. As alarms clear they are removed from the Current Alarm List. Pressing the RESET button compresses the remaining alarms. Computer alarms are also displayed on the Current Alarm List.

(3) Critical Function Monitor Page

The Critical Function Monitor Page displays each critical function, its alarm state and the presence
23 of a failed sensor in the alarm logic.

(4) Failed Sensor List

An alphanumeric alarm list which displays all input sensors which have failed out of range. Sensor point ID, English descriptor, time of failure and substituted value if appropriate are displayed. Indication is also made if the sensor is used for Critical Function Alarm purposes.

e. Level II display pages include:

(1) Core Systems Display

(2) Primary Systems Display

(3) Secondary Systems Display

(4) Containment Systems Display

f. Level III display pages include:

(1) Main Steam System Display

(2) Feedwater System Display

(3) Emergency Feedwater System Display

(4) Letdown Charging Display (CVCS-1)

(5) Boric Acid System Display (CVCS-2)

(6) Pressurizer Display

(7) Safety Injection System Display

(8) Containment Heat Removal Systems Display

(9) Containment Isolation Systems Display

(10) Shutdown Cooling System Display

g. Each CFMS display page is assigned a unique three digit page number. The first digit of the page number indicates the level of the display.

h. Movement through the display hierarchy is provided by using the PAGE, SECTOR, EXECUTE, FORWARD, and BACK keys on the CFMS alphanumeric keyboard.

i. Lateral movement through each level is provided by using the FORWARD and BACK keys. Each level page loop is closed so that paging forward from the last page moves the first page to the display screen.

j. Vertical movement between levels is made by using the SECTOR key followed by the sector number and the EXECUTE key. This key combination moves the display page associated with that sector to the screen. Each display page has up to nine available sectors. 23

k. When the SECTOR key alone is pressed, sector indicators are displayed which indicate all sectors available from the present display. Sector indicators are removed from the display after 30 seconds.

l. The key stroke combination of SECTOR and 0 returns the previous high level display to the screen. If the present page was reached by direct paging a default upper level page returns to the screen when SECTOR 0 is pressed.

m. Direct paging is provided to go directly to any page in the display hierarchy. Pressing the PAGE key followed by a three digit page number and the EXECUTE key displays the selected page.

n. Trend displays are accessed using a dedicated function button. One dedicated function button is provided for each trend page.

o. Error messages are provided for illegal page or sector commands. Error messages are removed after 5 seconds.

4. <u>Trend Displays</u>

a. Two trend display pages provide graphical, time based trends.

b. Each trend display page can trend up to four parameters (see figure III.A.1.2-3).

c. Parameters to be trended are assigned from the CFMS Programmers Console and cannot be reassigned from the CFMS User Display Station.

d. All four trends on a display page utilize the same time scale. Each trend display page may have a different time scale.

e. Alphanumeric annotation is provided on each trend page.
23 This includes scales, axes, and labels.

f. Each trend page is displayed by pressing a function key which provides direct access to the trend. Trend pages are not sectored to but may be accessed by direct paging. The trend function buttons may be activated at any time. Return from the trend pages is provided by a RETURN FROM TREND button which restores the page that was active when a trend was requested.

5. <u>Alarms</u>

The CFMS alarms annunciate specific information which is of importance to the user (see figure III.A.1.2-4).

a. Five classifications of alarms are provided:

(1) Critical Function Alarms

(2) Parameter Alarms

(3) Sector Alarms

(4) Failed Sensor Alarms

(5) Computer Failure Alarms

b. Critical Function Alarms

(1) Critical function alarms are provided on the Level I Critical Functions Monitor Page. These alarms are generated whenever alarm logic indicates that a critical function is not being maintained.

(2) Critical function alarms are only set when the required input signals are of good quality.

(3) Critical function alarms are only cleared when the required input signals are of good quality.

(4) Critical function alarm logic consists of software algorithms composed of arithmetic, comparison and Boolean capability.

(5) Critical function alarms provide an output signal to go to a supplied audible annunciator.

(6) Critical function alarms are annunciated by color change and blink.

(7) Critical function alarms retain the alarm color following acknowledgement. The audible alarm and blink are suppressed following acknowledgement.

(8) Critical function alarms are acknowledged by the 23
ACKNOWLEDGE button on the primary display station in the control room. Only this one display station is capable of acknowledging alarms. The ACKNOWLEDGE button only acknowledges CFMS alarms and does not affect other control room alarms.

(9) Critical function alarm status is displayed on all display pages using a three by three annunciator matrix (see figure III.A.1.2-5).

(10) A visual indication of a failed sensor input to the critical function alarm logic is provided on each page.

(11) Critical function alarms displayed on the Level I page include time and an English description of the alarm (see figure III.A.1.2-6).

c. Parameter Alarms

(1) Parameter alarms are provided whenever a parameter exceeds a High-High, High, Lo, Lo-Lo setpoint.

(2) High-High, High, Lo, and Lo-Lo setpoints with deadbands are required for each parameter alarm.

(3) Parameter alarms are annunciated by color change and blink.

(4) Parameter alarms are not audibly annunciated.

(5) Parameter alarm blink is supressed when the ACKNOWLEDGE button is pressed.

(6) All Parameter Alarms are listed on the Current Alarm List.

(7) The Current Alarm list includes:

(a) Name - Alphanumeric description of alarm

(b) Current Value - the currently measured value of the process parameter in engineering units

23 (c) Setpoint - the alarm violated setpoint

(d) Severity - the severity indicator of the alarm (Hi-Hi, Hi, Lo, Lo-Lo)

d. Sector Alarms

(1) Sector alarms are provided to alert the user that important changes have occurred on a lower level page. Up to nine sector alarms may appear on each page except Level III pages.

(2) Sector alarms blink until acknowledged. Sector alarms remain lit until cleared. Sector alarms are not audibly annunciated.

(3) Sector alarms are displayed in yellow when alarmed.

e. Failed Sensor Alarms

(1) In order to ensure a high availability system a failed sensor alarm is generated when a FAILED HIGH OUT OF RANGE or FAILED LOW OUT OF RANGE sensor is recognized. The failed sensor alarm message is written at the bottom of each page and remains until cleared. Each time a failed sensor is recognized the audible alarm output is activated and the alarm message blinked until acknowledged.

(2) All failed sensors are listed on the failed sensor list.

(3) Out-of-range setpoints are required to alarm the failed sensors.

f. Computer Alarms

Alarm outputs indicating computer malfunction are provided. These alarms generate an annunciator window and audible output signal. An alarm shall be provided to indicate failure of the CFMS. The presence of a computer alarm is indicated on every display page.

6. User Error Messages

a. User error messages are displayed on the bottom of each display as appropriate. User error messages remain on the display for 5 seconds and are then removed (see figure III.A.1.2-5).

b. User error messages are generated by improper page or sector requests. 23

2.2 HARDWARE DESIGN BASES

A. The CFMS hardware shall have sufficient calculational and memory capacity to support the functional requirements delineated in section 2.1.

B. The CFMS hardware shall have sufficient hardware features to support the functional requirements delineated in section 2.1.

C. The CFMS input hardware shall be capable of measuring a minimum of 200 plant process signals. Analog signals shall be measured with an overall accuracy of 0.25%.

D. The CFMS hardware shall be capable of providing output to:

1. Four independent CRT display stations

2. Analog output for strip chart recorder

3. Digital output for alarm annunciation

E. The CFMS shall be capable of operator interaction in the following manner:

1. Operator's Keyboard

   The operator shall have the capability of interacting with the CFMS through the use of a keyboard. The operator can perform the following functions:

   a. Display select

   b. Alarm acknowledge

   c. Alarm reset

   d. Trend selection

2. System Console

   The system console provides for control and monitoring of internal computer system operation. It is used by technicians to monitor the system's operation, to perform periodic testing, and to initiate software loading and maintenance.

F. The CFMS hardware shall utilize sufficient peripherals to support:

23

1. CFMS functional requirements delineated in section 2.1.

2. Sufficient bulk storage for:

   a. Program load function

   b. Historical data storage functions

   c. Online software functions

   d. Offline software maintenance

G. The CFMS shall utilize the following design techniques to achieve high reliability:

1. Minimize reliance on rotating memory for online operation

2. Burn-in of central processor and main memory at elevated temperature

3. Utilize an uninterruptible source of a-c power

4. Utilize on-line diagnostics to minimize MTTR

H. HARDWARE DESCRIPTION

A block diagram of the configuration is provided in figure III.A.1.2-7.

1. User Interface Stations

a. User interface stations shall be provided. Each user interface station shall consist of:

(1) A high resolution color CRT monitor

(2) An alphanumeric keyboard

(3) Function Keys to include:

- EXECUTE
- SECTOR
- PAGE
- FORWARD
- BACK
- ACKNOWLEDGE
- TREND 1
- TREND 2
- RETURN FROM TREND
- CURRENT ALARM LIST COMPRESSION (RESET) 23

(4) Numeric Key Pad

Numeric keypad and function keys must be arranged in a fashion which is consistent with human factors considerations.

2. Computer

a. Computing Capability

Computing capability shall be provided which will allow:

(1) Update of CRT displays

(2) Online conversion from plant signals to engineering units

(3) Offline data base maintenance

(4) Calculation of alarm algorithms

(5) Generation of 8-color graphical CRT displays

(6) Graphical video trending

(7) Online diagnostics

(8) Real time clock

(9) Watch dog timer - failure alarm

b. Programmer's Console

A programmer's console shall be provided which will allow:

Administrative control of access via password, key switch or comparable means

Interactive alphanumeric access to the CFMS computer

c. Storage

Sufficient storage shall be provided to allow:

23 (1) The CFMS shall be capable of storing the values of a minimum of 200 plant process signals for a minimum of 1 hour at 5-second increments. The values shall be time tagged.

(2) Storage and support of 20 displays

(3) Storage and support of CFMS data base

(4) Storage and support of required operating system software, etc.

(5) Storage and support of historic trend displays

(6) System load and support utilities

d. Input/Output Capability

Sufficient input/output capability shall be provided to support a minimum of 200 process signal inputs.

(1) Alarm digital outputs - The following contact outputs are provided.

(a) CFMS Alarm

(b) CFMS failure alarm

(c) CFMS failed sensor alarm

(2) Four analog outputs - Four analog outputs are provided for hard copy recorders. The parameters which are output are selected from the programmer's console.

(3) Conversion of signals to digital representations suitable for computer processing.

(4) The CFMS shall be capable of providing a simultaneous trend of up to four analog parameters. Analog outputs shall also be provided to allow capability for sine chart recording.

2.3 SOFTWARE DESIGN BASES

A. The software for the CFMS shall be designed utilizing a modular, top-down design approach whenever practical.

B. The software for the CFMS shall be thoroughly documented including:

1. Functional specification

2. Software specification

3. Program listings 23

C. The software shall be subjected to:

1. Verification by design review at each step in the design process in accordance with a verification plan

2. Validation by test in accordance with a validation plan

D. The CFMS software shall provide the implementation of the functional requirements of section 2.1.

E. The CFMS software shall have the capacity to be maintained in the following modes:

1. Calibration - Offline

The CFMS software shall support calibration of the process sensors including:

a. Engineering unit conversion constants

b. Alarm setpoints

2. Source Code Maintenance

The CFMS shall be capable of providing source code maintenance including the following utility program:

a. Editor

b. Debugger

c. Assembler/Compiler

d. Loader

F. ONLINE CAPABILITIES

The CFMS software shall be capable of providing:

1. Online User interaction

2. Online engineering units conversion

3. Input validity checking

4. Video data trending

5. Alarm generation, acknowledge and reset

23 6. Calculational capability

7. Online CFMS diagnostic and checking capability

8. Online trend point select and time increment select

9. Bootstrap loader

G. OFFLINE CAPABILITIES

The CFMS software shall be capable of providing:

1. Offline data base maintenance and editing. An offline interactive input maintenance facility shall be provided which allows:

a. Initial input, allocation and error checking of CFMS input data base files

b. Modification, addition, and deletion of points in the CFMS data base.

c. Report generation to provide formalized listings of the offline CFMS input data base files

2. Offline diagnostic and maintenance capabilities.

3. System generation facilities to be used to generate the CFMS software.

H. FAILURE CONSIDERATIONS

1. The CFMS shall utilize high reliability inputs and equipment but is not a Class 1E system.

2. Following recovery from a power failure to the CFMS, the system shall be capable of returning to an operational mode automatically.

3. No one out-of-range failed sensor shall cause the critical function alarms to give misleading indication.

4. The CFMS shall provide indicattions of:

   a. Sensor out-of-range failures

   b. Recoverable computer errors and exceptions

   c. Fatal computer errors

   d. Sensor out-of-range failures which affect the critical function alarm logic 23

2.4 HEALTH PHYSICS COMPUTER SYSTEM

2.4.1 OVERVIEW

The emergency assessment, radiological laboratory functions, and effluent management programs for the SONGS 2&3 require the integration of a complex and sophisticated system. This system should perform many computations based on a variety of input data and then output these results in various formats for the many tasks it performs.

The Health Physics System needs of SONGS stem from four primary types of requirements. The basic laboratory operations entail the support of various computer analysis, data storage, and report generation capabilities for the health physics and radiochemistry groups. The implementation of Appendix I to 10CFR50 via revision of the radiological effluents technical specifications (NUREG 0972) imposes additional computation, data storage, and reporting requirements, incorporating inputs from the site meteorological instrumentation. Requirements for post-accident preparedness have been imposed as a result of the accident at the Three Mile Island Plant (NUREG 0654 and Reg. Guide 1.97, Rev. 2). Data acquisition and analysis should be performed with the aid of various system radiation monitors and containment atmosphere monitors for Units 2&3. Readings from meteorological instrumentation, the radiation monitoring system, detectors in the field, and laboratory measurements should be assimilated to allow estimation of releases of radioactive materials and possible doses to offsite persons.

The central connection for these information processing requirements is the Health Physics Computer (Hewlett Packard 1000). The location for this computer is the Units 2&3 Counting Room. This unit will be a multi-task multi-user processor which will be capable of acquiring all necessary data either directly from the source or from intermediate processors or conditioners. This data will serve as the basis for the various computations which are necessary for the routine reports and also for the emergency assessment requirements.

There are basically two types of data which will be transmitted to the Health Physics Computer; analytical results and live-time[a] data. The analytical results will consist primarily of isotopic qualification and quantification of various system contents, i.e., liquid and gaseous effluents. Some of the analytical results may be transferable directly following review to the Health Physics Computer for incorporation into the system data base and subsequent computations. Other analyses may be entered manually from various plant locations. The locations for entry of analytical results will have data entry, system interrogation, and system feedback capability.

The second type of data to be transmitted to the Health Physics Computer is live-time digital data from various monitors throughout the system. The analogue signals generated from these monitors will be converted to digital signals and then transmitted to the Health Physics Computer System. This data will then be used in various routine and emergency computations as
23 well as being available for trend analysis and reports.

Various types of communication other than data receipt will take place in the Health Physics Computer System. These communications may be simple, or as complex as to being the primary computational center for offsite dose calculations.

There are several basic communication links in the site-wide system. The most straight forward are the digital data link and the multiplex A/D data links. The High Pressure Ion Chamber (HPIC) System will consist of inputs from various off-site locations which will be transmitting data on a routine basis and will also be capable of being interrogated by the Health Physics Computer at the discretion of the SONGS staff.

Of a similar nature is the multiplex A/D data from the various monitoring systems. The Radiation Monitoring System and the Meteorological Tower(s) data will be routinely transmitted to the Health Physics Computer system for use in various computations and system trending.

The ability to interface with the Technical Support Center is essential for the Health Physics Computer System. This system will provide the radiological portion of the emergency assessment function of the TSC system. The Health Physics System will function as a peripheral device available upon request to produce and transmit various sets of data or computational results.

a. This data may actually be averages for specific time periods.

2.4.2 SPECIFIC TECHNICAL CONDITIONS

A. The function of the Health Physics Computer System (HPCS) includes both routine and emergency support of the Health Physics Task. During routine operations, data acquisition systems will provide radiological and meteorological data to be incorporated into data bases for effluent calculations.

B. The emergency function of the Health Physics Computer System incorporates many of the routine data acquisition functions to be used in the Emergency Assessment and Response System. The basic information required to assess doses to individuals from emergency releases or during accident conditions describes the source term, the transport mechanism, and the receptors. The source term is a designation of the rate of release of each species of radioisotope at each moment in time throughout the release. Input data required for the source term description came from the RMS, flow monitors, calibration data files, accident modeling data files, and keyboard entries.

C. The effects of atmospheric dispersion in transportation of radioactive materials from the release point will be calculated from the measured meteorological parameters monitored by the SONGS meteorological tower. Meteorological data will also be available to the control room meteorological information system, the counting room processors, the Technical Support Center, as well as the Emergency Operation Facility. 23

E. Cumulated dose is of importance during and following an accidental release, as well as instantaneous dose rates. During the monitoring of the release, whole body dose, skin dose, and dose to the thyroid can be accumulated for a hypothetical individual at various distances in each sector through the duration of the accident.

F. Information describing location of the plume with respect to population and evacuation routes is of primary interest to a state agency, which has the responsibility for directing the relocation of members of the general public. Changing parameters, which are measured at the site, are used locally to perform calculations describing the plume and isodose curves.

G. An effective presentation of the best available description of current conditions may thus be made available at separated locations, allowing all parties to review and make decisions based on the same data. Hard copy printing of the graphically displayed information may be obtained at any time for records.

H. The primary locations which will be receiving this data are the Technical Support Center and the Emergency Offsite Facility.

I. The HPCS utilizes the various data input devices, then manages the data and performs various computations. The results of the HPCS computations and certain blocks of data which are maintained within the HPCS data bases will be transmitted to certain remote locations for display and/or review. In particular, the offsite emergency facilities, the onsite Technical Support Center, and the Control Room Meteorological Information Systems. Routine operation of these systems will assure operability on an emergency basis.

2.4.3 COMPUTERS

A. The Health Physics Computer System will be located within the SONGS 2&3 Counting Room. The system is comprised primarily of two Hewlett-Packard model 1000 processors and associated peripherials. These processors function in redundant capacities precluding the loss of function should one processor be scheduled for preventive maintenance or should one malfunction. The configuration of these processors provides dual mass storage maintenance as the system also incorporates two 120 Mbyte hard disk mass storage units. Identical records will be maintained to aid in data correlation to prevent the loss of either program or data correlation, and to prevent loss of either program or data essential to providing continuous operability. Each processor has the capability to access either disk during an operation should its primary disk fail for some reason. 23

B. A magnetic tape will perform transaction and data logging functions. This device might also be used as a supplemental mechanism for the incorporation of large data, such as historical meteorological data, into the systems data bases. This device will also serve as an archiving device for offline data storage.

C. The primary operator interactive devices include the CRT/terminals and printing devices within the SONGS 2/3 Counting Room, control room, Radiation Protection Cabinet TCC.

D. The HP-1000 system is configured with a "Power Fail Recovery System." This system provides battery sustaining power for memory during line power outages, as well as a battery charging circuit, and battery charge state testing. If a line power outage does not last long enough to deplete the available battery charge. the power fail/auto restart feature HP 1000 may be used to resume processing.

2.4.4 PROCESSORS

A. The Counting Room Systems for the SONGS 2&3 will support a range of data acquisition and analysis, data storage, and report generation activities. Gamma spectroscopy, utilizing a germanium detector, multi-channel analyzer, and computer for isotopic analysis, is required to analyze routine plant process and effluent sampliers for radionuclide composition. The same activity must be supported during the post-accident scenario, with the instrumentation of the unaffected unit serving as the onsite counting facility required by NUREG-0578.

B. The processor selected for use in these systems in the Nuclear Data Model ND6685 system. These units will be redundant to the extent of function and operation such that operators may utilize either system for analysis should the need arise. The routine applications for these systems will revolve to a great extent around compliance with various regulations and guidances from various regulatory agencies. During normal operations, the accountability for radioactive effluents in routine releases will be provided by these systems. Results of the grab sample analysis are integrated with other inputs and retained to form the data base from which reports are generated to satisfy requirements of Regulatory Guide 1.21. The respective counting room processors will monitor compliance with limiting conditions of operation in radiological effluents technical specifications, including various regulatory limits specified in 10CFR20 and 10CFR50. This monitoring function will be based on calculations of the values of limiting parameters in comparison with the regulatory limits.

C. In addition to the effluent functions mentioned above, certain routine process and reactor coolant sample analyses will be performed using these counting room systems. There also will be routine gamma spectroscopy analyses performed on routine health 23
physics samples such as contamination surveys, air samples, etc. The processor chosen for this application is basically identical to the counting room processors previously mentioned.

2.4.5 DATA ACQUISITION UNITS

A. As a result of a detailed evaluation of both the SONGS specific configuration and known parameter monitoring, a requirement for a system to acquire certain data and transmit it for incorporation into data bases and calculations was realized. These data signals are typically analog signals from the plant Radiation Monitoring System (RMS) or other systems. The Data Acquisition Unit (DAU) will receive these signals after interrogating the monitor, then digitize the signal, and transmit the digital signal to both the Health Physics Computer and to the associated counting room processor. The basis of this redundancy is assurance of data availability during normal and emergency modes of operation as well as assurance of operability of the emergency data transfer link on a continuous basis.

B. During normal operations, this data will be utilized by the counting room processors for incorporation into the routine effluent management program as well as other activities. Certain tasks such as trending effluent monitors during releases and monitor setpoint determination will be performed based on this data acquired by the DAU's. In order to accurately track the rates of release of routine radioactive effluents to the environment and to estimate the resulting possible offsite dose, readings of effluent radiation monitors and the corresponding flow readings must be monitored on a continuous basis.

C. Emergency releases require more frequent monitoring due to the transient nature of release concentrations in the post-accident case. Each reading must be correlated in time with concurrent flow data and meteorological conditions to afford the prediction of the dispersion and transport of radioactive materials from the site. Certain area and process radiation monitor readings may provide information as to the location of plant problems, integrity of radiation barriers, and the results of mitigative actions in the post-accident scenario.

D. The DAU's proposed for the San Onofre application consists of a Nuclear Data acquisition system Model ND-68DC. This unit will receive the signals, digitize them, then transmit the digital signals to the Counting Room Processors and the Health Physics Computer. This acquisition system will have internal programming capabilities sufficient to schedule the data acquisition and perform all the tasks necessary to complete the function. The list of parameter signals to be received and processed by the Data Acquisition Units are shown in Table III.A.1.2-2.

23 2.4.6 METEOROLOGICAL INFORMATION CENTER

The control room complex will be equipped with a meteorological information system. This system consists of a Hewlett-Packard 2624P terminal printer and CRT unit. The control room operator will have the capability to review current meteorological data or obtain historical information. Also available is the capability to obtain a hard copy of any display the operator feels need documentation.

The HP 2624P interfaces directly to the HPCS. This is the source of the meteorological data as the met towers transmit data to the HPCS periodically. This data is to be incorporated into a data base for use with the counting room processor as well as the control room meteorological information system.

The meteorological information to be input to the HPCS is listed in Table III.A.1.2-3.

2.4.7 TECHNICAL SUPPORT CENTER

NUREG-0578 and NUREG-0696 both address the function and configuration of an onsite Technical Support Center. The Health Physics Computer System will support the TSC during emergency modes of operation by providing displays and computational results to aid in the determination of various radiological aspects of plant activities during the course of an accident. The devices which support these activities consist of a Hewlett-Packard 2648A graphics terminal, a HP-2631G graphics printer, and a remote monitor of the terminal display.

2.4.8 POST ACCIDENT SAMPLE SYSTEM

The data which will be obtained from this system will be used primarily during the post accident evaluation of reactor coolant and containment atmosphere status.

The data will be transmitted to the Units 2&3 counting room processors. This will then be incorporated into the data which will be available for transmission to the Health Physics Computer and there be available to the various locations of overall emergency response and evaluation system.

2.4.9 PRESSURIZED ION CHAMBERS

Radiation detectors will be displayed in the external evironment at SONGS. These detectors in an accident situation, would allow a direct assessment of offsite dose and a check on calculations of offsite dose based on site radiation measurements and meteorological data. This confirmation is particularly important because of the complex meteorological conditions due to the land-ocean interface and adjacent surface terrain features. These detectors will also indicate the location of the radiation plume in order to allow the appropriate field sampling.

2.4.10 EMERGENCY OPERATIONS FACILITY 23

This will be the offsite communication of emergency information including radiological release roles, meteorological measurements and dose projections.

2.4.11 FUTURE CAPABILITIES

A. Interface with the SCE corporate IBM computer.

B. Additional meteorological towers.

C. Expansion to include displays in the Emergency Operations Facility.

D. The NRC data link.

E. The Office of Emergency Services link.

F. Personnel Exposure Support System.

2.4.12 SEISMIC

The H.P. computer system is not safety related, and is non-seismic.

2.4.13 POWER SUPPLY

Uninterruptible power for the counting room computers and peripherals is provided by a dedicated UPS system consisting of a 125 VDC battery, with
23 90 minutes capacity, battery circuit breaker, 125 VDC battery charger powered from a Class IE source, inverter, and a.c. distribution panel. Back-up power supply is provided by a non-Class IE source, step-down transformer, and regulating transformer. A static transfer switch is provided to transfer the power supply from the normal source to the back-up source in the event of failure of the main power source.

Table III.A.1.2-1
DATA POINTS DISPLAYED IN THE CFMS (Sheet 1 of 7)

Tag No.	Point Identification	Type of Input	
ZL84212	ATMOS DUMP VLV SG1	CI	
ZL84191	ATMOS DUMP VLV SG2	CI	
PT4703	AUX FD PUMP DISC PRESS(P140)	ANA-I	
PT4710	AUX FD PUMP DISC PRESS(P141)	ANA-I	
ZT4713	AUX FW CNTRL VLV (TO SG1)	CI	
ZT4706	AUX FW CNTRL VLV (TO SG1)	CI	
ZT4712	AUX FW CNTRL VLV (TO SG2)	CI	
ZT4705	AUX FW CNTRL VLV (TO SG2)	CI	
FT4725	AUX W FLOWRATE TO SG1	ANA-I	
FT4720	AUX FW FLOWRATE TO SG2	ANA-I	
ZL47311	AUX FW ISO VLV (TO SG1)	CI	
ZL47152	AUX FW ISO VLV (TO SG1)	CI	
ZL47302	AUX FW ISO VLV (TO SG2)	CI	
ZL47141	ZUX FW ISO VLV (TO SG2)	CI	
YS4707	AUX FW PUMP P141 STATUS	CI	
YS4733	AUX FW PUMP P504 STATUS		
ZH4716	AUX FW TURB STOP VLV	CI	
ZL92011	AUX SPRAY ISO VLV	CI	
ZH92471	BA MU TO CH PMP SUCT VLV	CI	23
ZL9257	BA MU TO CH PMP X-CONN VLV	CI	
ZL0201B	BACKPRESS CONTROL VLV	CI	
ZL0201A	BACKPRESS CONTROL VLV	CI	
YS0100F	BACKUP HEATER STATUS 1	CI	
YS0100E	BACKUP HEATER STATUS 2	CI	
YS0100D	BACKUP HEATER STATUS 3	CI	
YS0100G	BACKUP HEATER STATUS 4	CI	
YS0100H	BACKUP HEATER STATUS 5	CI	
YS0100I	BACKUP HEATER STATUS 6	CI	
ZL40532	BLOWDOWN ISO VLV SG2	CI	
ZL40541	BLOWDOWN ISO VLV SG2	CI	
ZH92352	BOR ACID MAKEUP TK A GRAV FD	CI	
ZH92402	BOR ACID MAKEUP TK B GRAV FD	CI	
ZL0210Y	BORIC ACID FLOW CONTROL VLV	CI	
F210Y	BORIC ACID FLOWRATE	ANA-I	
YS92411	BORIC ACID MAKEUP PMP A STAT	CI	
YS92421	BORIC ACID MAKEUP PMP B STAT	CI	
L206A	BORIC ACID MAKEUP TK A LVL	ANA-I	
L208A	BORIC ACID MAKEUP TK B LVL	ANA-I	
A203	BORONOMETER	ANA-I	
XCCAS1	CCAS STATUS 1	CI	
XCCAS2	CCAS STATUS 2	CI	
ZL62112	CCW INLET VLV	CI	
ZL62162	CCW OUTLET VLV	CI	
XCEDM1	CEDM POWER SUP UNDVOLT REL	CI	

Table III.A.1.2-1
DATA POINTS DISPLAYED IN THE CFMS (Sheet 2 of 7)

23

Tag No.	Point Identification	Type of Input
XCEDM2	CEDM POWER SUP UNDVOLT REL	CI
P212	CHARG PUMP DISCH HDR PRESS	ANA-I
P212	CHARGING FLOWRATE	ANA-I
ZH92001	CHARGING ISO VLV	CI
ZH9203	CHARGING ISO VLV LOOP 1A	CI
ZH9202	CHARGING ISO VLV LOOP 2A	CI
YS92281	CHARGING PUMP NO 1 STATUS	CI
YS92291	CHARGING PUMP NO 2 STATUS	CI
YS92292	CHARGING PUMP NO 2 STATUS	CI
YS92302	CHARGING PUMP NO 3 STATUS	CI
XCIAS1	CIAS STATUS 1	CI
XCIAS2	CIAS STATUS 2	CI
R17804	CNMT AIR BORNE RAD MONITOR	ANA-I
RM78572	CNMT DOME RAD MONITOR	ANA-I
TT99112	CNMT DOME TEMP	ANA-I
LT93891	CNMT EMER SUMP LEVEL	ANA-I
ZH93051	CNMT EMERG SUMP ISO VLV	CI
ZH93042	CNMT EMERG SUMP ISO VLV	CI
ZH93031	CNMT EMERG SUMP ISO VLV	CI
ZH93022	CNMT EMERG SUMP ISO VLV	CI
YS99531	CNMT FAN CLR E399 STATUS	CI
YS99392	CNMT FAN CLR E400 STATUS	CI
YS99471	CNMT FAN CLR E401 STATUS	CI
YS99552	CNMT FAN CLR E402 STATUS	CI
ZL99511	CNMT LG VOL PURGE DISCH VLV	CI
ZL99502	CNMT LG VOL PURGE DISCH VLV	CI
ZL99491	CNMT LG VOL PURGE INLET VLV	CI
ZL99482	CNMT LG VOL PURGE INLET VLV	CI
ZL98231	CNMT MINIPURGE INLET VLV	CI
ZL98212	CNMT MINIPURGE INLET VLV	CI
ZL98251	CNMT MINIPURGE OUTLET VLV	CI
ZL98242	CNMT MINIPURGE OUTLET VLV	CI
ZL99002	CNMT NORM A/C INLET	CI
ZL99201	CNMT NORM A/C INLET VLV	CI
ZL99712	CNMT NORM A/C OUTLET VLV	CI
ZL99211	CNMT NORM A/C OUTLET VLV	CI
P352A	CNMT PRESS WR	ANA-I
P351A	CNMT PRESS WR	ANA-I
P303X	CNMT SPRAY 1 PRESS	ANA-I
T303X	CNMT SPRAY 1 TEMP	ANA-I
P303Y	CONMT SPRAY 2 PRESS	ANA-I
T303Y	CNMT SPRAY 2 TEMP	ANA-I
ZH93671	CNMT SPRAY HDR 1 ISO VV	CI

Table III.A.1.2-1
DATA POINTS DISPLAYED IN THE CFMS (Sheet 3 of 7)

Tag No.	Point Identification	Type of Input
ZH93682	CNMT SPRAY HDR 2 ISO VV	CI
F338	CNMT SPRAY NO. 1 FLOWRATE	ANA-I
F348	CNMT SPRAY NO. 2 FLOWRATE	ANA-I
YS93951	CNMT SPRAY PUMP NO 1	CI
YS93962	CNMT SPRAY PUMP NO 2	CI
ZL58031	CNMT SUMP PUMP 1/C ISO VLV	CI
ZL58042	CNMT SUMP PUMP O/C ISO VLV	CI
T111Y	COLD LEG TEMP LOOP 1A	ANA-I
T115Y	COLD LEG TEMP LOOP 1B	ANA-I
T125Y	COLD LEG TEMP LOOP 2A	ANA-I
T121Y	COLD LEG TEMP LOOP 2B	ANA-I
RT7818	COND AIR EJECT RAD MON	ANA-I
YS3278	COND PUMP P050 STATUS	CI
YS2179	COND PUMP P051 STATUS	CI
YS3279	COND PUMP P052 STATUS	CI
YS3281	COND PUMP P053 STATUS	CI
PT3202	COND VACUUM LP ZONE N	ANA-I
PT3395	COND VACUUM LP ZONE S	ANA-I
LT3293	CONDENSATE STORAGE TK LEVEL	ANA-I
XCP1S1	CPIS STATUS 1	CI
XCP1S2	CPIS STATUS 2	CI
XCSAS1	CSAS STATUS 1	CI
XCSAS2	CSAS STATUS 2	CI
XEFAS1	EFAS STATUS #1	CI
XEFAS2	EFAS STATUS #2	CI
ZH63701	EMERG CCW INLET ISO VLV	CI
ZH63722	EMERG CCW INLET ISO VLV	CI
ZH63661	EMERG CCW INLET ISO VLV	CI
ZH63682	EMERG CCW INLET ISO VLV	CI
ZH63711	EMERG CCW OUTLET ISO VLV	CI
ZH63732	EMERG CCW OUTLET ISO VLV	CI
ZH63671	EMERG CCW OUTLET ISO VLV	CI
ZH63692	EMERG CCW OUTLET ISO VLV	CI
FT1111	FW FLOWRATE TO SG1	ANA-I
FT1121	FW FLOWRATE TO SG2	ANA-I
ZL40522	FW ISO VLV (TO SG1)	CI
ZL40481	FW ISO VLV (TO SG2)	CI
PT4014	FW PUMP P062 DISCH PRESS	ANA-I
YS4522	FW PUMP P062 STATUS	CI
PT4015	FW PUMP P063 DISCH PRESS	ANA-I
YS4523	FW PUMP P063 STATUS	CI
TE3922	FW TEMP TO STM GEN NO. 1	ANA-I
TE3921	FW TEMP TO STM GEN NO. 2	ANA-I
T111X	HOT LEG TEMP LOOP 1	ANA-I

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Table III.A.1.2-1
DATA POINTS DISPLAYED IN THE CFMS (Sheet 4 of 7)

Tag No.	Point Identification	Type of Input
T2121X	HOT LEG TEMP LOOP 2	ANA-I
F311	HPSI COLD LEG 1A FLOWRATE	ANA-I
F321	HPSI COLD LEG 1B FLOWRATE	ANA-I
F331	HPSI COLD LEG 2A FLOWRATE	ANA-I
F341	HPSI COLD LEG 2B FLOWRATE	ANA-I
ZH93241	HPSI HDR 1 TO 1A AI INJ VLV	CI
ZH93271	HPSI HDR 1 TO 1B INJ VLV	CI
ZH93301	HPSI HDR 1 TO 2A INJ VLV	CI
ZH93331	HPSI HDR 1 TO 2B INJ VLV	CI
ZH93232	HPSI HDR 2 TO 1A INJ VLV	CI
ZH93262	HPSI HDR 2 TO 1B INJ VLV	CI
ZH93292	HPSI HDR 2 TO 2A INJ VLV	CI
ZH93321	HPSI HDR 2 TO 2B INJ VLV	CI
ZH94201	HPSI HDR1 HOTLEG INJ VLV	CI
ZH94342	HPSI HDR2 HOTLEG INJ VLV	CI
F391	HPSI HOT LEG 1 FLOWRATE	ANA-I
F390	HPSI HOT LEG 2 FLOWRATE	ANA-I
P308	HPSI PUMP HDR #1 PRESS	ANA-I
23 P309	HPSI PUMP HDR #2 PRESS	ANA-I
YS93921	HPSI PUMP P017 STATUS	CI
YS93932	HPSI PUMP P018 STATUS	CI
YS93942	HPSI PUMP P019 STATUS	CI
ZH0224B	ION EXCH BYPASS VLV	CI
ZL0224B	ION EXCH BYPASS VLV	CI
ZL02211	LD TO REGEN HX CONTROL VLV	CI
ZL92042	LD TO REGEN HX CONTROL VLV	CI
P201	LETDN HX DISCH PRESS	ANA-I
T224	LETDN HX DISCH TEMP	ANA-I
ZL0110B	LETDOWN CONTROL VLV	CI
ZL0110A	LETDOWN CONTROL VLV	CI
F202	LETDOWN FLOWRATE	ANA-I
ZL92051	LETDOWN ISO VLV	CI
ZL92672	LETDOWN ISO VLV	CI
T351Y	LPSI HDR TEMP	ANA-I
F306	LPSI HEADER FLOWRATE	ANA-I
YS93901	LPSI PUMP P015 STATUS	CI
YS93912	LPSI PUMP P016 STATUS	CI
ZH93222	LPSI TO 1A INJ VLV	CI
ZH93251	LPSI TO 1B INJ VLV	CI
ZH93281	LPSI TO 2A INJ VLV	CI
ZH93312	LPSI TO 2B INJ VLV	CI
ZT1105	MAIN FW BYPASS (SG1) VLV	CI
ZT1106	MAIN FW BYPASS (SG2) VLV	CI
ZT1111	MAIN FW CNTRL VLV POS (SG1)	CI

Table III.A.1.2-1
DATA POINTS DISPLAYED IN THE CFMS (Sheet 5 of 7)

Tag No.	Point Identification	Type of Input
ZT1121	MAIN FW CNTRL VLV POS (SG2)	CI
ZH4047	MAIN FW STOP VLV (TO SG1)	CI
ZH4051	MAIN FW STOP VLV (TO SG2)	CI
ZL82011	MAIN STM TO AUX FWPT ISO SG1	CI
ZL82002	MAIN STM TO AUX FWPT ISO SG1	CI
XMSIS1	MSIS STATUS 1	CI
XMS1S2	MSIS STATUS 2	CI
ZL82042	MISV (SG1)	CI
ZL82051	MSIV (SG2)	CI
ZL82022	MSIV BYPASS VLV SG1	CI
ZL82031	MSIV BYPASS VLV SG2	CI
ZL92531	MU TO VCT VLV	CI
XZPDIL	PDIL DEVIATION ALARM	CI
RT7808	PLANT VENT STACK MONITOR	ANA-I
F210X	PMW FLOWRATE	ANA-I
ZL0210X	PMW FLOWRATE CONTROL VLV	CI
YS7149	PMW PUMP (P200) STATUS	CI
YS7150	PMW PUMP (P201) STATUS	CI
LT7133B	PMW TK LEVEL	ANA-I
R202	PROCESS RAD MONITOR	ANA-I
YS9170	PROP HEATER STATUS 1	CI
YS9171	PROP HEATER STATUS 2	CI
L110X	PZR LVL CH X	ANA-I
L110Y	PZR LVL CH Y	ANA-I
P100Y	PZR PRESS	ANA-I
P100X	PZR PRESS	ANA-I
T107	PZR RV RG200 DISCH TEMP	ANA-I
T108	PZR RV RC201 DISCH TEMP	ANA-I
ZL0100A	PZR SPRAY FCV	CI
ZL0100B	PZR SPRAY FCV	CI
T101	PZR WATER TEMP	ANA-I
L116	QUENCH TANK LEVEL	ANA-I
P116	QUENCH TANK PRESS	ANA-I
T116	QUENCH TANK TEMP	ANA-I
RT7813	RADWASTE DISCH LINE RAD MON	ANA-I
XRAS1	RAS STATUS 1	CI
XRAS2	RAS STATUS 2	CI
ZL75121	RC DRAIN TANK PUMP DISCH ISO	CI
ZL75132	RC DRAIN TANK PUMP DISCH ISO	CI
Y19160A	RCP 1A AMPS	ANA-I
YS9160A	RCP 1A STATUS	CI
Y19161A	RCP 1B AMPS	ANA-I
YS9161A	RCP 1B STATUS	CI

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Table III.A.1.2-1
DATA POINTS DISPLAYED IN THE CFMS (Sheet 6 of 7)

23

Tag No.	Point Identification	Type of Input
Y19162A	RCP 2A AMPS	ANA-I
YS9162A	RCP 2A STATUS	CI
YI9163A	RCP 2B AMPS	ANA-I
YS9163A	RCP 2B STATUS	CI
ZH93001	REFUEL WTR TK ISO VLV EAST	CI
ZH93012	REFUEL WTR TK ISO VLV WEST	CI
L302	REFUELING WATER TANK LEVEL	ANA-I
L301	REFUELING WATER TANK LEVEL	ANA-I
T221	REGEN HX DISCH TEMP	ANA-I
ZH0227C	RWT/BA INLET VLV	CI
ZH93402	SAFETY INJ TK NO 1 ISO VLV	CI
ZH93501	SAFETY INJ TK NO 2 ISO VLV	CI
ZH93601	SAFETY INJ TK NO 3 ISO VLV	CI
ZH93702	SAFETY INJ TK NO 4 ISO VLV	CI
L311	SAFETY INJECTION TK #1 LEVEL	ANA-I
P311	SAFETY INJECTION TK #1 PRESS	ANA-I
L321	SAFETY INJECTION TK #2 LEVEL	ANA-I
P321	SAFETY INJECTION TK #2 PRESS	ANA-I
L331	SAFETY INJECTION TK #3 LEVEL	ANA-I
P331	SAFETY INJECTION TK #3 PRESS	ANA-I
L341	SAFETY INJECTION TK #4 LEVEL	ANA-I
P341	SAFETY INJECTION TK #4 PRESS	ANA-I
ZL0306	SDC FLOWRATE CONTROL VLV	CI
ZL93784	SDC ISO VLV (I/C)	CI
ZL93773	SDC ISO VLV (I/C)	CI
ZL93392	SDC ISO VLV (I/C)	CI
ZL93371	SDC ISO VLV (I/C)	CI
ZL93791	SDC ISO VLV (O/C)	CI
ZL93362	SDC ISO VLV (O/C)	CI
ZL9316	SDC STOP VLV	CI
ZL79112	SERV WTR TO CNMT SUMP ISO	CI
LT1114	SG1 LEVEL (WIDE RNG)	ANA-I
FT1011	SG1 STEAM FLOWRATE	ANA-I
LT1124	SG2 LEVEL (WIDE RNG)	ANA-I
FT1021	SG2 STEAM FLOWRATE	ANA-I
ZL93341	SI DRAIN/TEST VLV	CI
XSIAS1	SIAS STATUS 1	CI
XSIAS2	SIAS STATUS 2	CI
XCPSSR	STARTUP CHANNEL X COUNTRATE	CI
P1013A	STEAM GEN #1 PRESS CH A	ANA-I
P1023A	STEAM GEN #2 PRESS CH A	ANA-I
T105	SURGE LINE TEMP	ANA-I
ZL8423	TURBINE BYPASS VLV	CI
ZL8426	TURBINE BYPASS VLV	CI

Table III.A.1.2-1
DATA POINTS DISPLAYED IN THE CFMS (Sheet 7 of 7)

Tag No.	Point Identification	Type of Input
ZL8425	TURBINE BYPASS VLV	CI
ZL8424	TURBINE BYPASS VLV	CI
RT7821	TURBINE PLANT SUMP RAD MON	ANA-I
PH2200A1	TURBINE STOP VLV	CI
PH2200D2	TURBINE STOP VLV	CI
PH2200E4	TURBINE STOP VLV	CI
PH2200H3	TURBINE STOP VLV	CI
ZL022732	VCT DISCH ISO VLV	CI
L226	VCT LEVEL	ANA-I
P225	VCT PRESS	ANA-I
T225	VCT TEMP	ANA-I
ZH0227A	VCT/RADWASTE SELECT VLV	CI
ZL0227A	VCT/RADWASTE SELECT VLV	CI
XCPSWR	WIDE RANGE POWER (CH A)	CI
PT4736	AUX FW PUMP DISC PRESS (P504)	ANA-I
TT6472	EMERG CCW TEMP LOOP A	ANA-I
TT6473	EMERG CCW TEMP LOOP B	ANA-I
FT6277	EMERG CCW TEMP LOOP A	ANA-I
FT6329	EMERG CCW TEMP LOOP B	ANA-I

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Table III.A.1.2-2
PARAMETERS TO BE INPUT INTO THE HP COMPUTER SYSTEM
(Sheet 1 of 3)

SONGS 2&3

	Title	Tag No.
1.	Emergency Radiation Monitoring System	2RE-7858-1
2.	Emergency Radiation Monitoring System	2RE-7859-2
3.	Emergency Radiation Monitoring System	2RE-7860-3
4.	Emergency Radiation Monitoring System	3RE-7858-1
5.	Emergency Radiation Monitoring System	3RE-7859-2
6.	Emergency Radiation Monitoring System	3RE-7860-3
7.	Containment Area Monitor	2RE-7856-1
8.	Containment Area Monitor	2RE-7857-2
9.	Containment Area Monitor	3RE-7856-1
10.	Containment Area Monitor	3RE-7857-2
11.	Containment Area Monitor	2RE-7848
12.	Containment Area Monitor	3RE-7848
13.	Containment Area Monitor	2RE-7845
14.	Containment Area Monitor	3RE-7845
15.	Control Room Area Monitor	2/3RE-7851
16.	Safety Equipment Building Area Monitor	2RE-7847
17.	Safety Equipment Building Area Monitor	3RE-7847
18.	Radiochem Lab Area Monitor	2/3RE-7852
19.	Containment High Range Monitor	2RE-7820-1
20.	Containment High Range Monitor	2RE-7820-2
21.	Containment High Range Monitor	3RE-7820-1
22.	Containment High Range Monitor	3RE-7820-2

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Table III.A.1.2-2
PARAMETERS TO BE INPUT INTO THE HP COMPUTER SYSTEM
(Sheet 2 of 3)

SONGS 2&3

	Title	Tag No.
23.	Wide Range Effluent Monitor	2RE-7865-1
24.	Wide Range Effluent Monitor	3RE-7865-1
25.	Wide Range Condenser Air Ejector	2RE-7870-1
26.	Wide Range Condenser Air Ejector	3RE-7870-1
27.	Main Steam Line Monitors	2RE-7874A1
28.	Main Steam Line Monitors	2RE-7875A1
29.	Main Steam Line Monitors	2RE-7874B1
30.	Main Steam Line Monitors	2RE-7875B1
31.	Main Steam Line Monitors	3RE-7874A1
32.	Main Steam Line Monitors	3RE-7875A1
33.	Main Steam Line Monitors	3RE-7874B1
34.	Main Steam Line Monitors	3RE-7875B1
35.	Condenser Air Ejector	2RE-7818
36.	Condenser Air Ejector	3RE-7818
37A	Control Room Airborne Monitor	2/3RE-7824-1A
37B.	Control Room Airborne Monitor	2/3RE-7824-1B
38A.	Control Room Airborne Monitor	2/3RE-7825-2A
38B.	Control Room Airborne Monitor	2/3RE-7825-2B
39A.	Plant Vent Stack Monitor	2/3RE-7808A
39B.	Plant Vent Stack Monitor	2/3RE-7808B
39C.	Plant Vent Stack Monitor	2/3RE-7808C
40A.	Containment Airborne Monitor	2RE-7804-1A
40B.	Containment Airborne Monitor	2RE-7804-1B

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Table III.A.1.2-2
PARAMETERS TO BE INPUT INTO THE HP COMPUTER SYSTEM
(Sheet 3 of 3)

SONGS 2&3

	Title	Tag No.
40C.	Containment Airborne Monitor	2RE-7804-1C
41A.	Containment Airborne Monitor	2RE-7807-2A
41B.	Containment Airborne Monitor	2RE-7807-2B
41C.	Containment Airborne Monitor	2RE-7807-2C
42A.	Containment Airborne Monitor	3RE-7804-1A
42B.	Containment Airborne Monitor	3RE-7804-1B
42C.	Containment Airborne Monitor	3RE-7804-1C
43A.	Containment Airborne Monitor	3RE-7807-2A
43B.	Containment Airborne Monitor	3RE-7807-2B
43C.	Containment Airborne Monitor	3RE-7807-2C
44.	Radwaste Discharge Line Monitor	2/3RE-7813
45.	Waste Gas Header Monitor	2/3RE-7814
46.	Gas Decay Tank Discharge Rate	2/3FIT-7202
47.	Radwaste Discharge Rate	2/3FE-7643
48.	Main Steam from Steam Gen. Flow Rate	2FE-1011
49.	Main Steam from Steam Gen. Flow Rate	2FE-1021
50.	Main Steam from Steam Gen. Flow Rate	3FE-1011
51.	Main Steam from Steam Gen. Flow Rate	3FE-1021

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Table A.1.2-3

PARAMETERS TO BE INPUT INTO HP COMPUTER SYSTEM

METEOROLOGICAL Tower

1. Wind Speed	10 meters
2. Wind Direction	10 meters
3. Wind Speed	40 meters
4. Wind Direction	40 meters
5. Temperature	10 meters
6. Delta T	primary (10-40m)
7. Delta T	secondary (10-40m)
8. Sigma-Theta	10 meters
9. Dewpoint Temperature	10 meters
10. Precipitation	Ground level

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III.A.2 - NUREG-0737 IMPROVING LICENSEE EMERGENCY PREPAREDNESS--LONG TERM

REQUIREMENT

Position

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

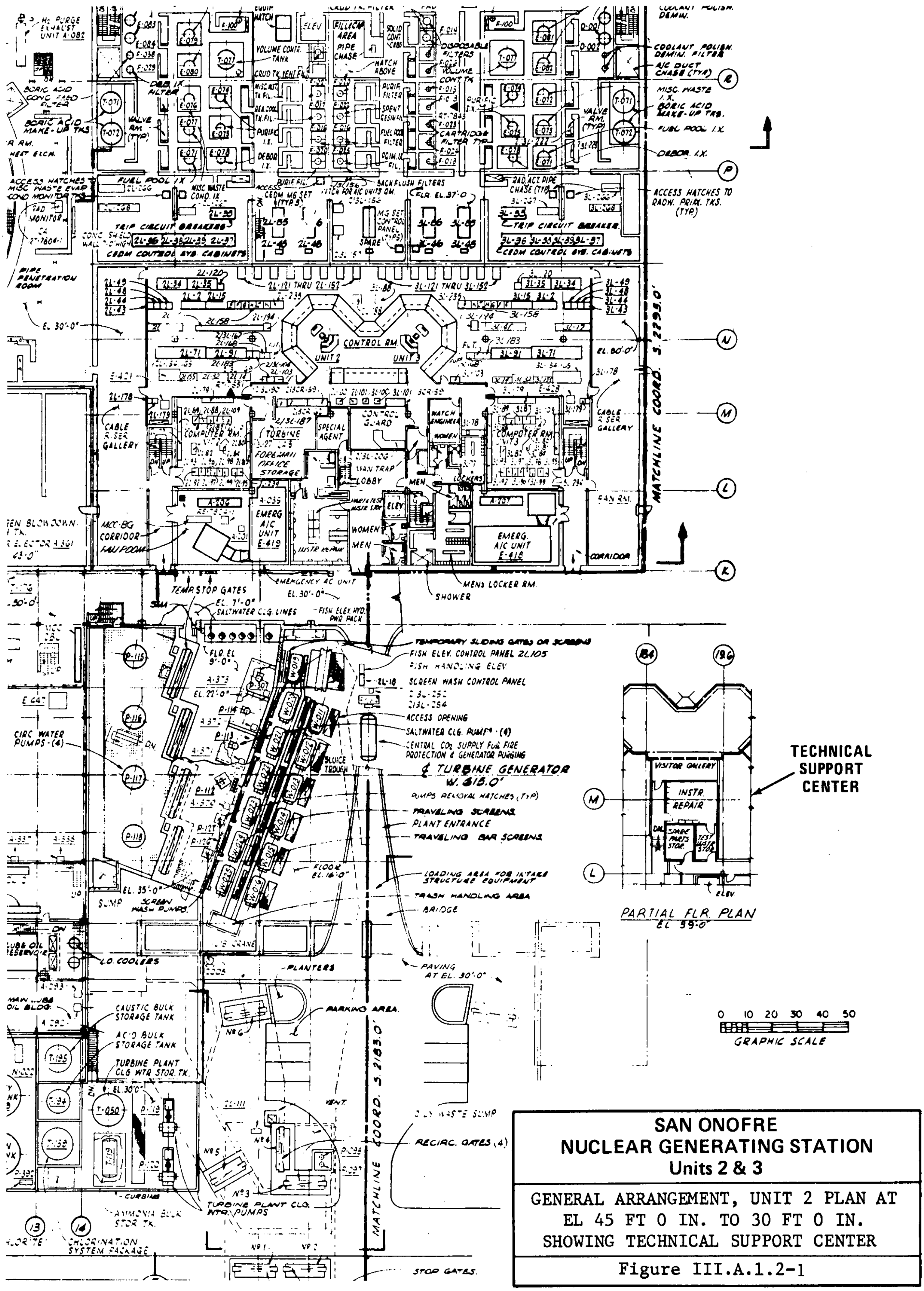
23 Response:

The applicants submitted a revised SONGS 2&3 Emergency Plan on August 18, 1980, which was written considering the amendments to 10 CFR Part 50, Appendix E, and NUREG-0654 (January 1980 for Interim Use and Comment).

Subsequent to that submitted, NRC issued NUREG-0654, Rev. 1. Both the applicants and NRC stuff are currently reviewing the SONGS 2&3 Emergency Plan to ensure compliance with NUREG-0654 Rev. 1. The applicants will revise the SONGS 2&3 Emergency Plan consistent with NRC clarification and in-house review prior to fuel load SONGS 1 (See III.A.1.1 and III.A.1.2).

Reference

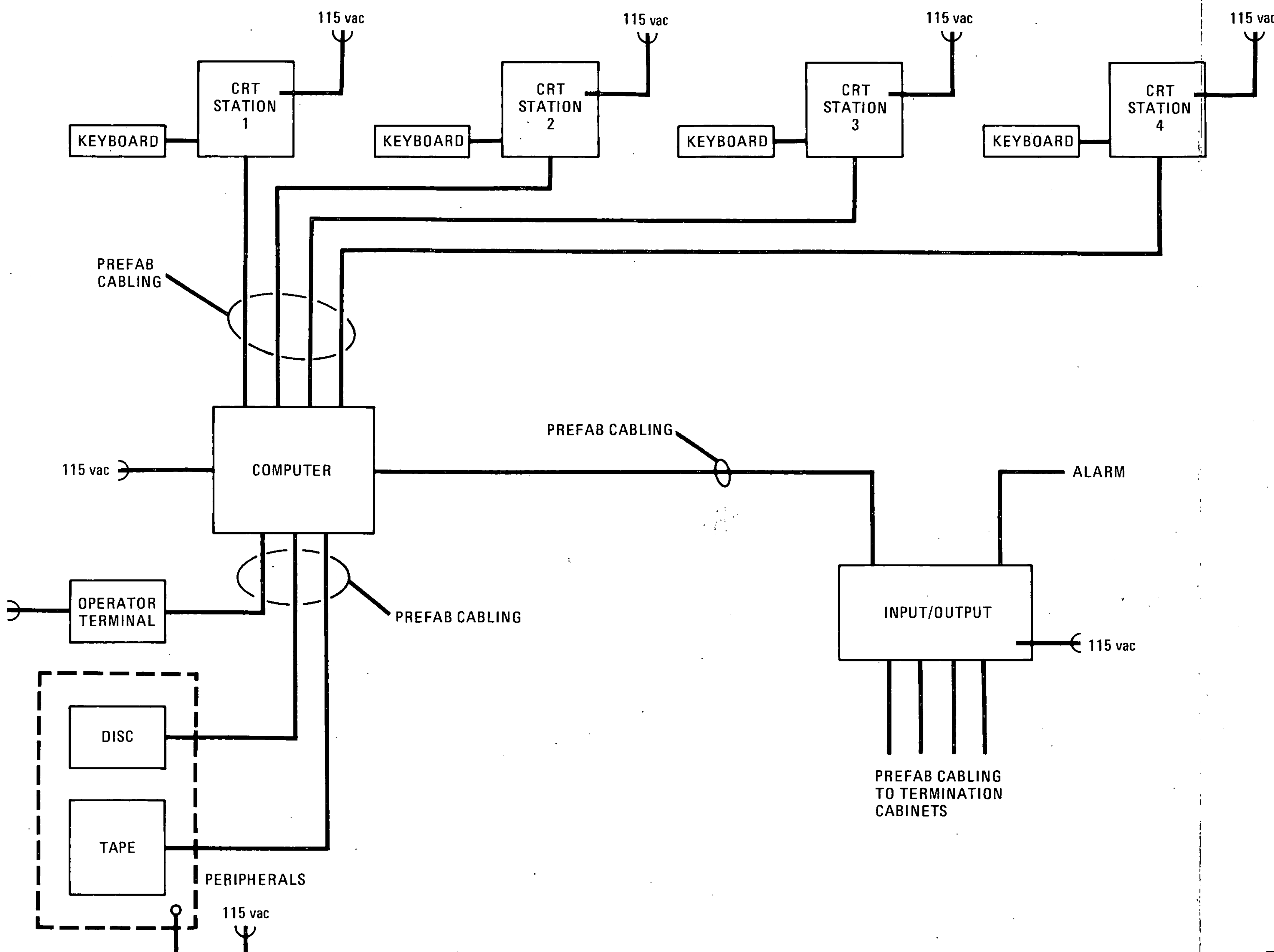
Response to NUREG-0660 item III.A.1.1 and III.A.1.2.



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

GENERAL ARRANGEMENT, UNIT 2 PLAN AT EL 45 FT 0 IN. TO 30 FT 0 IN. SHOWING TECHNICAL SUPPORT CENTER

Figure III.A.1.2-1



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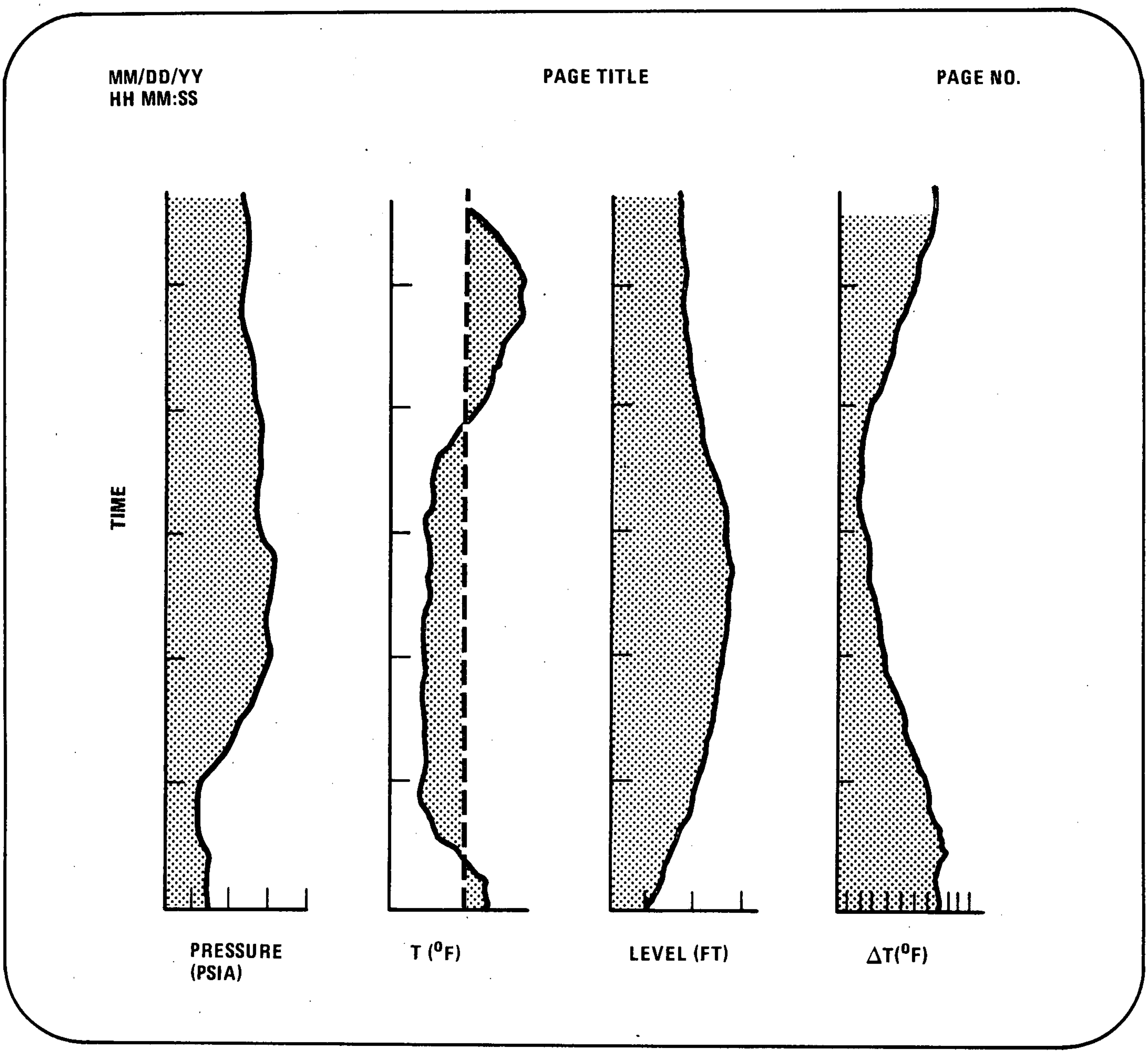
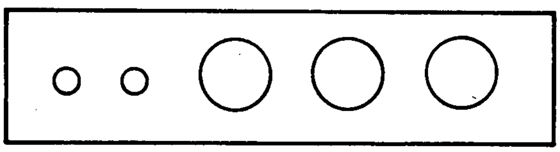
Units 2 & 3

CRITICAL FUNCTION MONITORING SYSTEM CONFIGURATION

Figure III.A.1.2-2





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TYPICAL TREND DISPLAY

Figure III.A.1.2-3

ALARM	BEFORE ACKNOWLEDGE: AUDIBLE OUTPUT	BEFORE ACKNOWLEDGE: SYMBOL BLINK	BEFORE ACKNOWLEDGE: COLOR CHANGE	AFTER ACKNOWLEDGE: AUDIBLE OUTPUT	AFTER ACKNOWLEDGE: SYMBOL BLINK	AFTER ACKNOWLEDGE: COLOR CHANGE
CRITICAL FUNCTION ALARM	Y	Y	Y	N	N	Y
PARAMETER ALARM	N	Y	Y	N	N	Y
SECTOR ALARM	N	Y	Y	N	N	Y
FAILED SENSOR	Y	Y	Y	N	N	Y
COMPUTER FAILURE	Y	–	–	N	–	–

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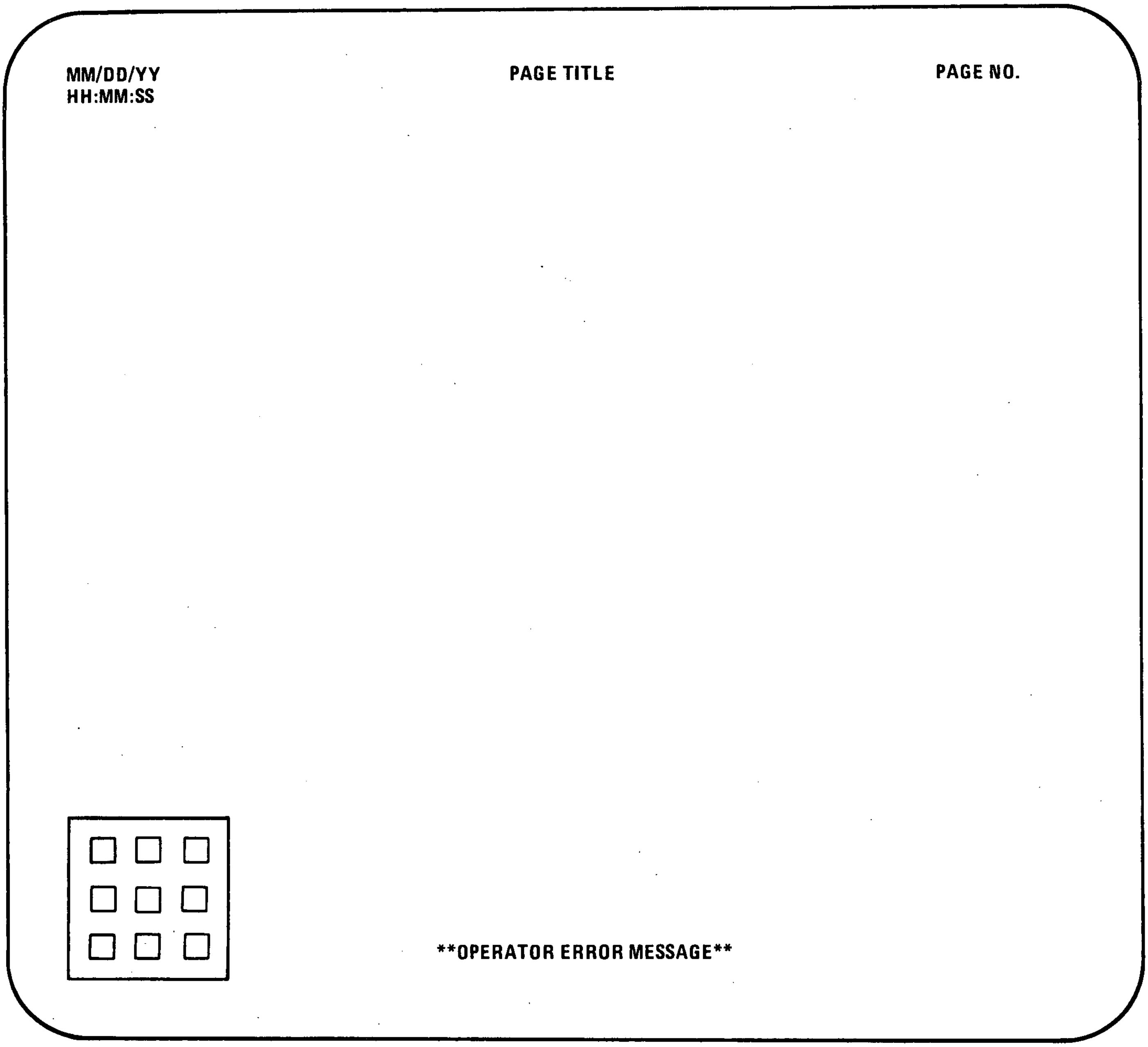
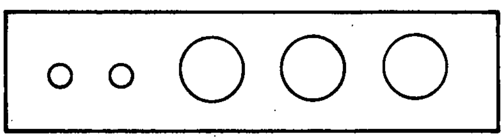
ALARM CHARACTERISTICS

Figure III.A.1.2-4





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Units 2 & 3

TYPICAL PAGE LAYOUT

Figure III.A.1.2-5

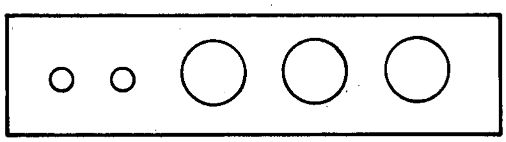
MM/DD/YY
HHMM: SS

CRITICAL FUNCTION ALARM MONITOR

PAGE NO.

FUNCTION	TIME ALARMED
CORE REACTIVITY CONTROL	HHMM:SS
CORE HEAT REMOVAL	
RCS INVENTORY CONTROL	
RCS PRESSURE CONTROL	
RCS HEAT REMOVAL	**FAILED**
CONTAINMENT PRESSURE/TEMP CONTROL	
CONTAINMENT ISOLATION	

FAILED SENSORS



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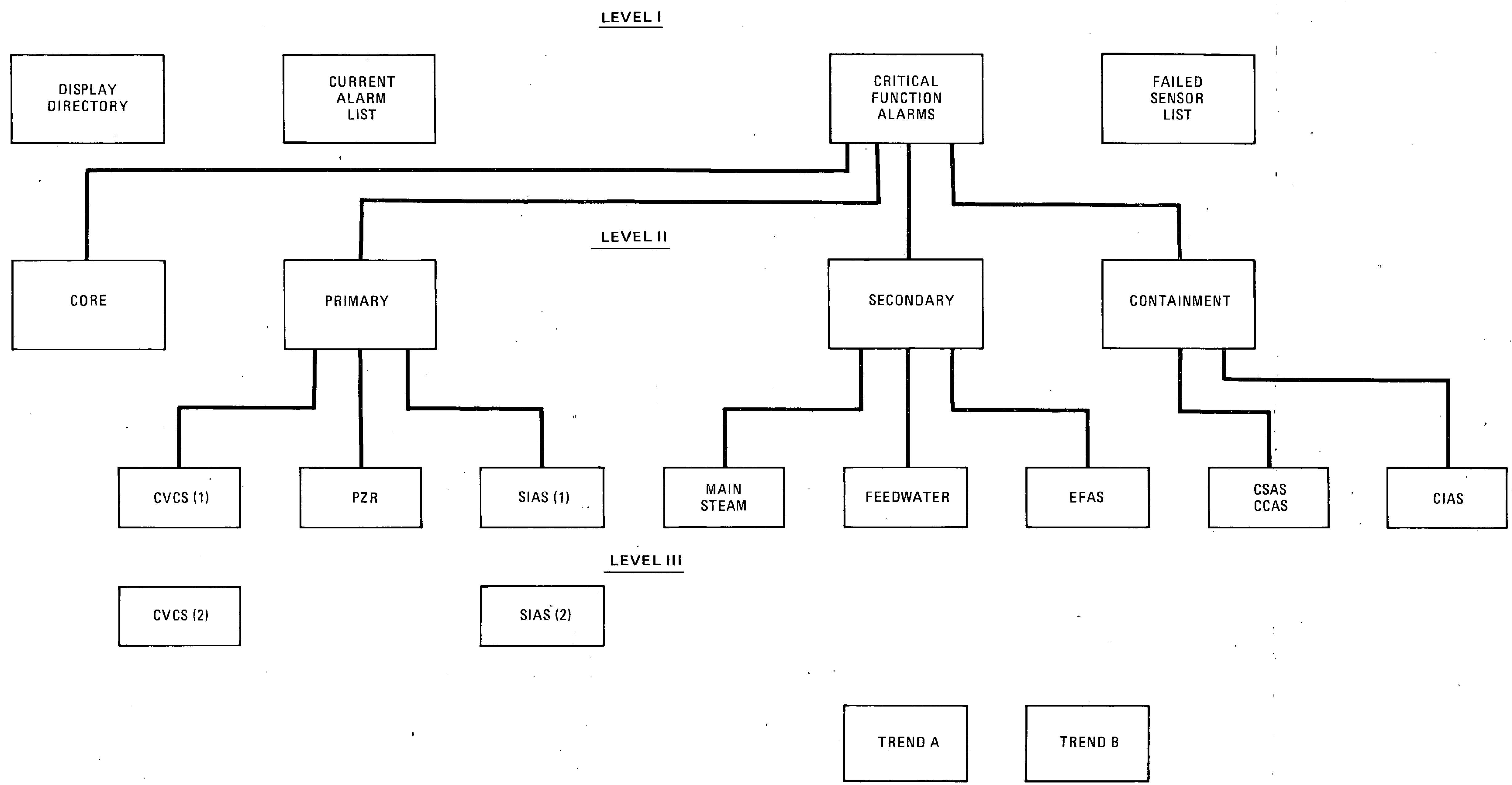
Units 2 & 3

TYPICAL CFMS ALARM PAGE

Figure III.A.1.2-6



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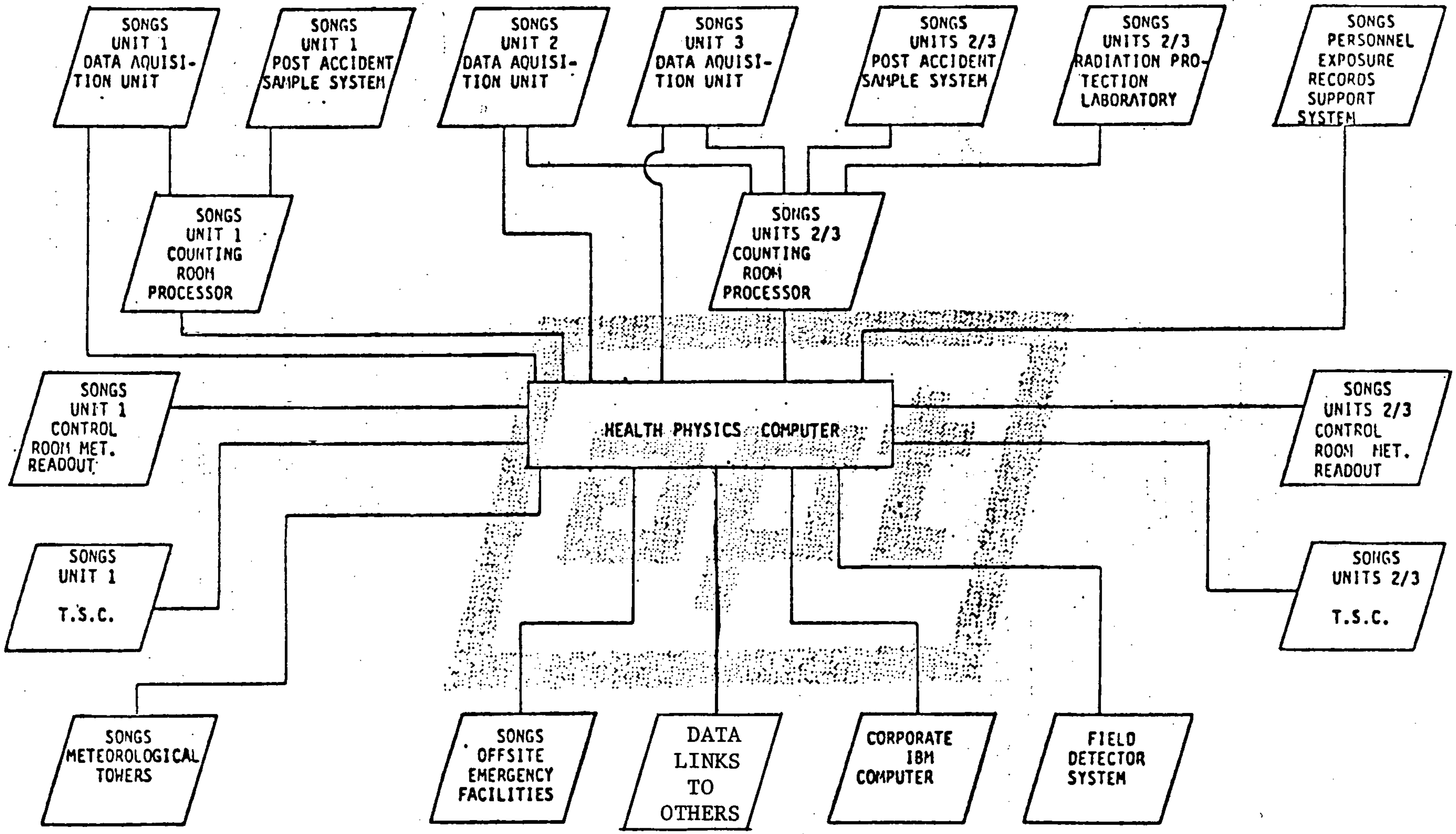
Units 2 & 3

TYPICAL DISPLAY HIERARCHY

Figure III.A.1.2-7



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Units 2 & 3

SONGS HEALTH PHYSICS DATA TRANSFER SYSTEM

Figure III.A.1.2-8

III.D.1.1 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED-WATER REACTORS AND BOILING-WATER REACTORS

REQUIREMENT

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate leak reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction - Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle. 23

Clarification

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

1. Systems that should be leak tested are as follows (any other plant system which has similar functions or postaccident characteristics even though not specified herein, should be included):

   - Residual heat removal (RHR)
   - Containment spray recirculation
   - High-pressure injection recirculation
   - Containment and primary coolant sampling
   - Reactor core isolation cooling
   - Makeup and letdown (PWRs only)

- Waste gas (includes headers and cover gas system outside of containment in addition to decay or storage system)

  Include a list of systems containing radioactive materials which are excluded from program and provide justification for exclusion.

2. Testing of gaseous systems should include helium leak detection or equivalent testing methods.

3. Should consider program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to all operating nuclear power plants regarding North Anna and related incidents, dated October 17, 1979.

RESPONSE

SCE has instituted a program to maintain leakage rates of systems outside containment to as low as practical which consists of the following:

Systems included in the program: (We have reviewed plant systems and identified the following systems outside containment that could potentially contain highly radioactive fluids following a serious accident. See figures III.D.1.1-1 through III.D.1.1-4.)

1. High pressure safety injection system (recirculation portion only).

23

2. Low pressure safety injection system (shutdown coding portion only).
3. Reactor coolant sampling system. (Post-accident sampling piping only.)
4. Containment spray system.
5. Radioactive waste gas system. (Post-accident sampling return piping only.)
6. Liquid radwaste system. (Post-accident sampling return piping only.)

Systems excluded from the program: (They will not preclude any option of cooling the reactor core nor will they prevent the use of needed safety systems.)

1. Radioactive liquid waste system except as discussed above (excluded by NRC in regional meeting).
2. Radioactive waste gas system except as discussed above. (The system is not required for use post-accident. The reactor coolant system degassing post-accident is accommodated by the reactor coolant gas vent system discussed in II.B.1.)

3. Reactor coolant letdown system. (The system is not required to function post-accident. The plant can be brought to a cold shutdown condition without the letdown system. The letdown system is isolated on SIAS and CIAS.)

4. Reactor coolant pump seal bleed-off system. (The system is not required to function outside containment post-accident. The seal bleed-off system is isolated outside containment on CIAS. The system remains isolated post-accident. If seal bleed-off is required post-accident, pressure in the seal bleed-off header will increase and the header relief valve will lift providing a flow path to the quench tank.)

5. Charging System. (The charging system under post-accident conditions does not contain radioactive fluid since the letdown system is isolated as discussed in item 3 above. The charging system takes suction from the refueling water storage tank.)

A. Immediate leak reduction measures:

1. All vent and drain lines will be capped to prevent release due to seal leakage.

2. The packing of all valves (except Kerotest which is a packless, stainless steel diaphragm valve) in the scoped liquid systems will be inspected for leakage or evidence of leakage such as 23 boric acid accumulation. Maintenance will be performed on the packing of liquid system valves identified as requiring work.

3. The seals and packing on all pumps in the scoped liquid systems will be inspected for leakage or signs of leakage.

4. Valves, fittings, and compressor seals in the scoped gaseous systems will be "snooped" for leakage. Maintenance will be performed on gas system valves and instrument fittings identified during leak tests as requiring work.

B. Procedures for determining (measuring) leakage:

During the inspections described above, leakage rates will be recorded in drops per minute or hour. After completion of the inspection, the following additional leak rate tests will be performed:

1. High pressure safety injection system - integrated leak rate test.

2. Low pressure safety injection system - integrated leak rate test.

3. Charging system - integrated leak rate test.

4. Reactor coolant sampling system - integrated leak rate test.

5. Containment spray system - integrated leak rate test.

6. Radioactive waste gas system - bubble ("snoop") test of individual valves, fittings, and seals. Quantitative value obtained from bubble rate.

7. Liquid Radwaste system - integrated leak rate test.

C. Leakage measurement results:

Actual leakage rates will be determined during plant startup testing prior to initial criticality and reported to the NRC.

D. Continuing leak reduction:

A general station procedure will be written incorporating the applicable inspections performed during the leak reduction program described above.

The leak rate measurements will be performed at intervals not to exceed each refueling outage. Records of leakage rates will be retained in the station files.

E. The potential release path identified in NRC letter dated October 17, 1979, re: Radioactive Release at North Anna Unit 1 and Lessons Learned is not credible in the SONGS 2&3 design. High level in the
23 volume control tank automatically diverts the letdown liquid to the coolant radwaste system. These tanks provide 240,000 gallons of hold-up capacity. In addition, high level alarms are provided on the hold-up tanks. The tanks have surface diaphrams to minimize off-gasing and tritium releases due to evaporation. An overall program for the prevention of unplanned release of radioactivity was implemented in response to I.E. Circular 79-21. A summary is provided below.

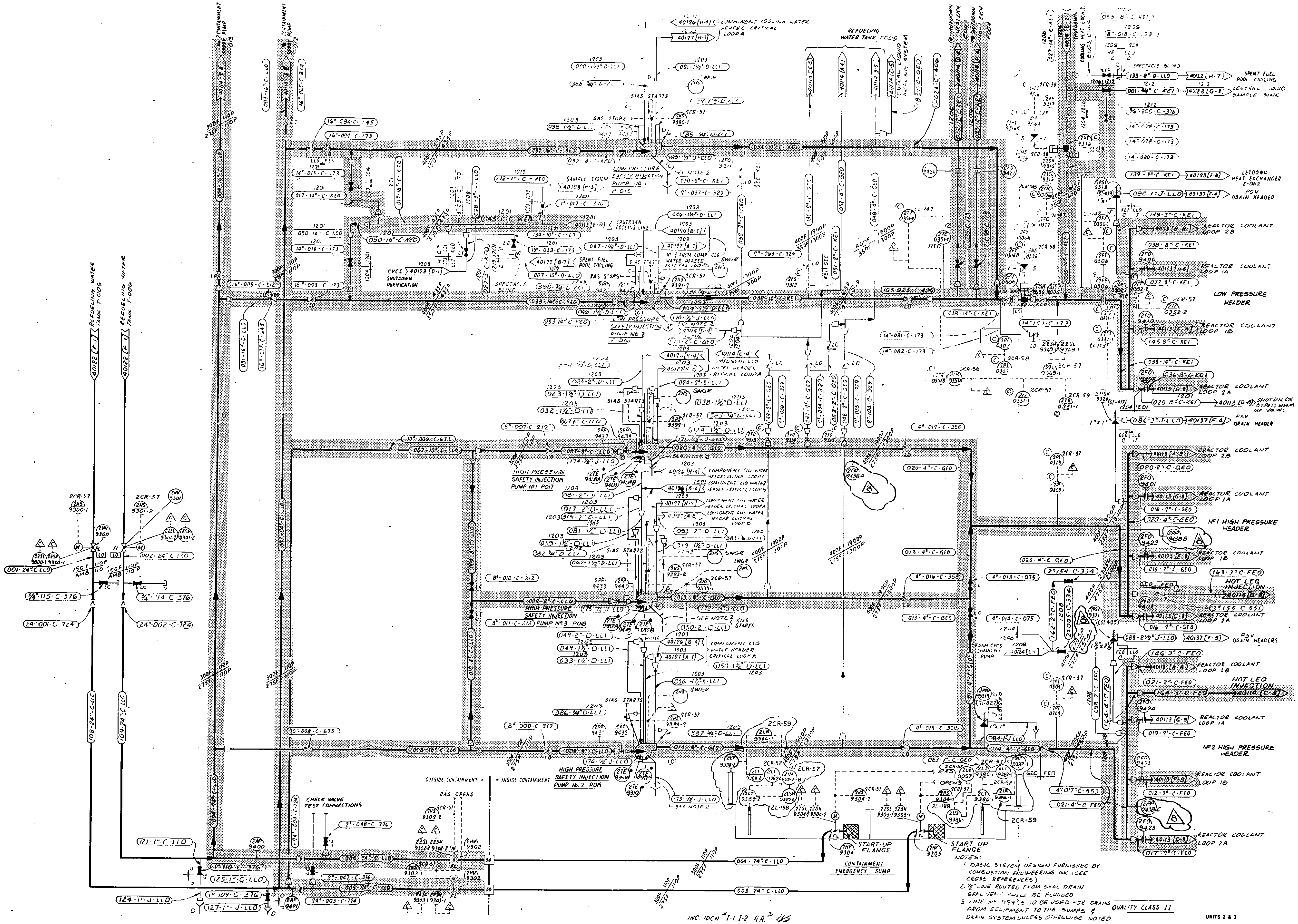
1. All tanks containing radioactive liquid are located inside buildings and the overflows are routed to the liquid radwaste system via the radwaste building sump. Furthermore, all tanks have level indication, and all tanks, with the exception of the RWST, have high level alarms to alert control room personnel.

2. Storm drains are located outside the plant and there are no existing crossconnects between floor drains or sumps and the storm drains.

3. All pumps that contain radioactive liquid are equipped with drip pans that are hard piped to the floor drains. All valves that contain radioactive liquid are located in valve galleries that have sloping floors leading to floor drains.

4. All doors that have the possibility of radioactive water leaking under them are watertight or equipped with cofferdams.

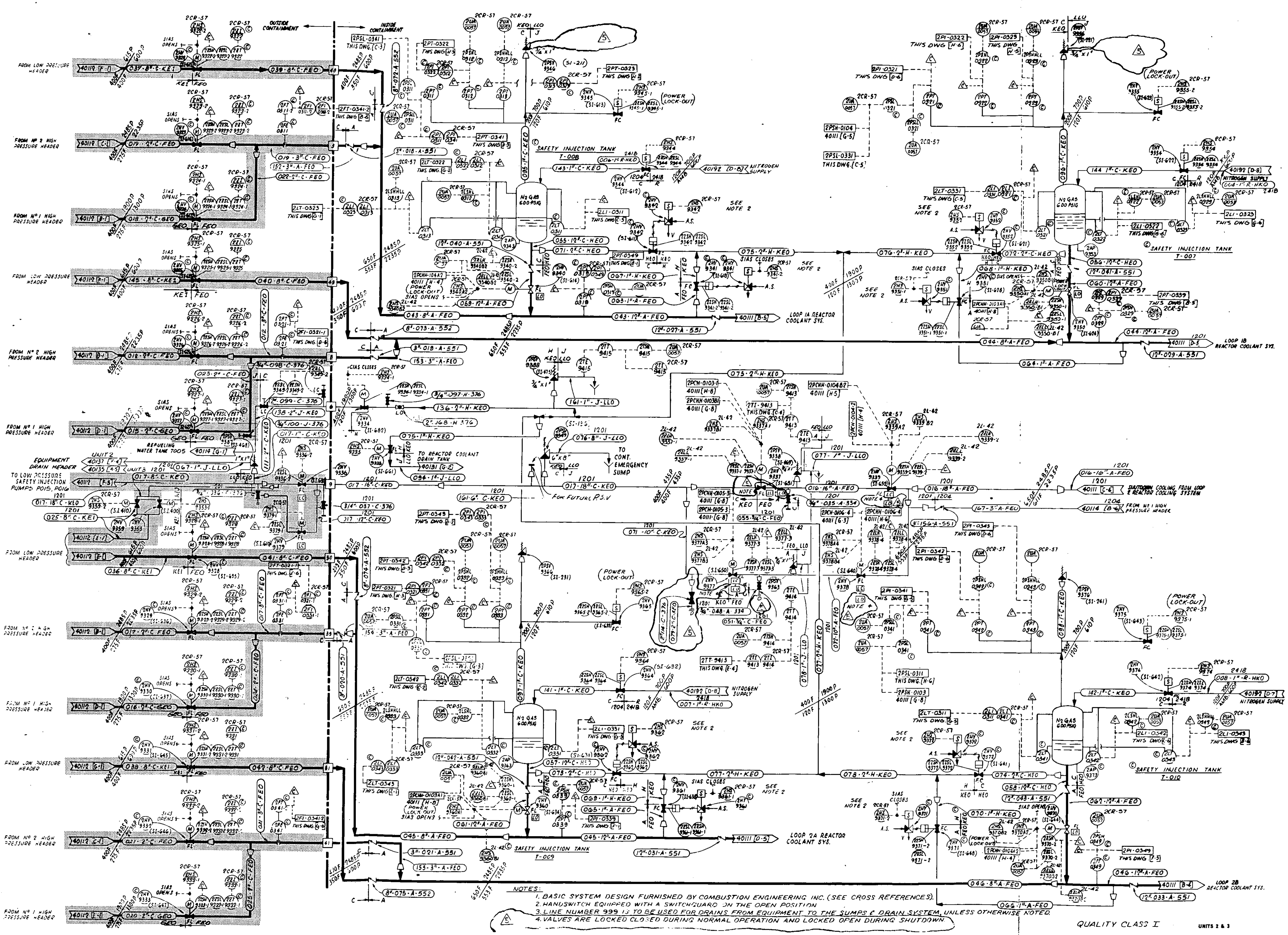
5. All crossconnects between units 2 & 3 were reviewed, and it was determined that at the time of Unit 2 startup, all crossconnects will be locked closed, capped, or flanged. 23



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

PRELIMINARY SCOPING FOR LEAKAGE TESTING OF THE SAFETY INJECTION SYSTEM

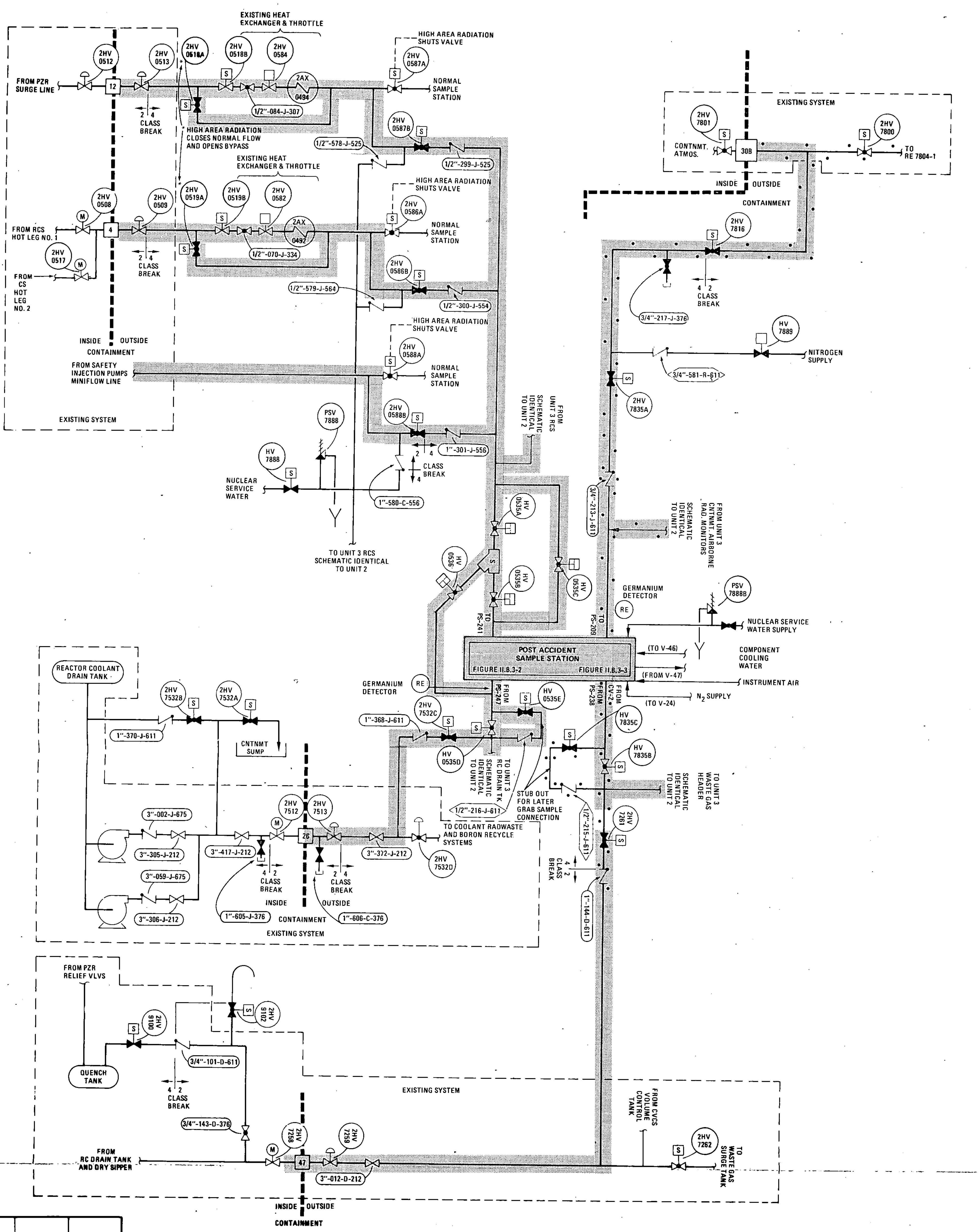
Figure III.D.1.1-1



SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

PRELIMINARY SCOPING FOR LEAKAGE TESTING OF THE SAFETY INJECTION SYSTEM

Figure III.D.1.1-2



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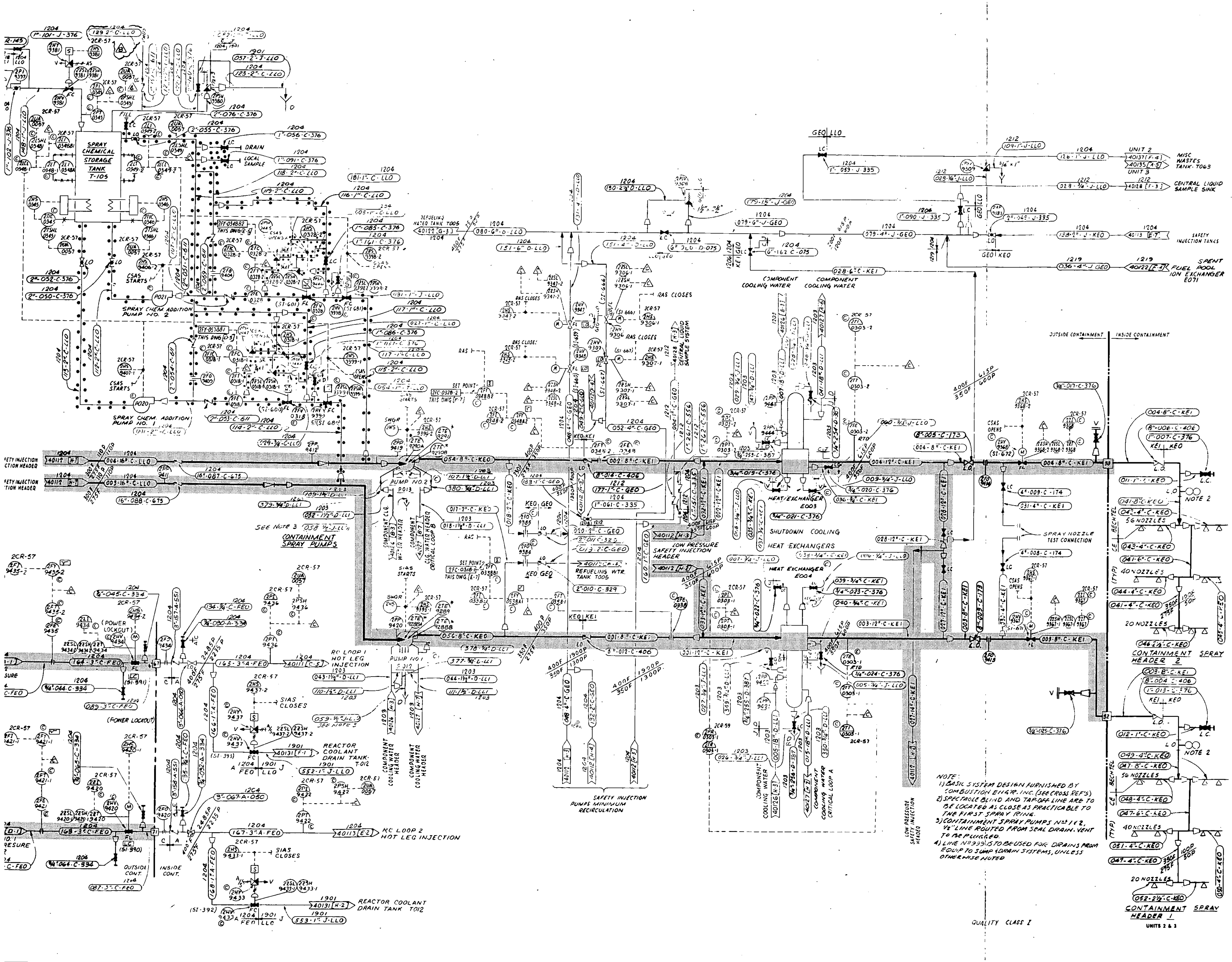
SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

PRELIMINARY SCOPING FOR LEAKAGE TESTING OF THE POST-ACCIDENT SAMPLING SYSTEM

Figure III.D.1.1-3

Amendment 23



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3

PRELIMINARY SCOPING FOR LEAKAGE TESTING OF THE CONTAINMENT SPRAY SYSTEM

Figure III.D.1.1-4

III.D.3.3 - NUREG 0660 INPLANT RADIATION MONITORING

REQUIREMENT

Position

1. Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident. 23

2. Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Clarification

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

1. The physical size of the auxiliary/fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

2. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

3. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

4. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high dose rate areas. 23

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

For applicants with fuel-loading dates prior to January 1, 1981, provide by fuel loading (until January 1, 1981) the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler

with attached single-channel analyzer (SCA). The SCA window should be calibrated to the 365 KeV of iodine-131 using the SCA. This will give an
23 initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

RESPONSE

1.1 GENERAL DESCRIPTION

Plant systems for monitoring inplant radiation and airborne radioactivity in the main control room are being modified to include instruments appropriate for a broad range of routine and emergency conditions. Effective monitoring of increasing iodine levels in plant areas other than the main control room, under accident conditions, will include the use of portable instruments. Calibration methods will be developed for both permanently installed and portable instruments.

1.2 DESIGN CRITERIA

The above instrumentation will be designed to meet the following criteria:

A. Instrumentation for monitoring inplant airborne radioactivity will have the capability to accurately detect the presence of iodine in the area of interest following an accident. This will be accomplished by using an iodine sampler with attached single channel analyzer (SCA). The sampler in the control room will be permanently installed and a portable or cart-mounted sampler will be used for areas outside of the control room. The SCA window will be calibrated to the 365 keV photon of ^{131}I. A representative air sample will be taken and then counted for ^{131}I using the SCA. This will give an initial conservative estimate of the presence of iodine and can be used to determine if respiratory protection is required. Care will be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

B. Capability will be provided to remove the sampling cartridge to a low background, low contamination area for further analysis. This area will be ventilated with clean air containing no significant quantities of airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Prior to counting, the sample will be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The capability will be provided to measure accurately the iodine concentrations present on these samples and effluent charcoal samples under accident conditions.

1.3 DETAILED DESCRIPTIONS

1.3.1 AIRBORNE MONITORING OF THE CONTROL ROOM

Existing airborne radiation instrumentation (RE-7824-1 and RE-7825-2), currently configured to detect the presence of radioactivity entering the control room at the start of an accident and to initiate the emergency ventilation system, will be modified so that it also meets post-accident monitoring requirements. Neither the skid-mounted sampler detector assembles nor their associated ratemeter/single channel analyzer modules require any modification. The sampler suction and discharge lines, however, will be changed to allow the monitors to sample the appropriate HVAC ducts both prior to and subsequent to the initiation of emergency ventilation system operation. RE-7824-1 will be provided with additional piping and valves and an additional isokinetic probe to allow it to draw from the emergency recirculation train A duct, upstream of filters. Manual realignment of the suction and discharge valves will be done by the operator. RE-7825-2 will be provided with sample piping extensions to sample ducting which is common to both the normal intake path and the emergency recirculation train B path upstream of filters. No additional isokinetic probe will be required for RE-7825-2.

1.3.2 AIRBORNE MONITORING OF AREAS OUTSIDE OF THE CONTROL ROOM

Airborne monitoring of areas outside of the control room will be accomplished through a combination of two portable cart-mounted SCA's (Technical Associates Model BAM 3IM) and battery-powered air samplers. 23
Charcoal or silver zeolite cartridges from the air samplers will be delivered to the radiochemistry laboratory for analysis in the multichannel analyzer.

1.3.3 LABORATORY ANALYSIS OF AIRBORNE RADIOACTIVITY SAMPLES

Laboratory analysis of airborne radioactivity samples will be accomplished by use of a 4000 channel gamma ray Ge(Li) multi-channel analyzer located in the counting room at elevation 63'-6 of the auxiliary building. As discussed in the responses to the shielding (II.B.2) and post-accident sampling (II.B.3) items, the counting room is expected to be a low background area. In the unlikely event background radiation is unacceptably high, the nearby Unit 1 counting room is available for analyzing Unit 2 and 3 samples. 23
Charcoal sample cartridges will be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. For silver zeolite sample cartridges, no purge is necessary.

1.4 MEETING NRC REQUIREMENTS

The area radiation and airborne radioactivity monitor instrumentation discussed in FSAR subsection 12.3.4, in conjunction with the currently proposed modifications to the inplant radiation monitoring system, fulfills the NRC requirements as outlined in NUREG-0578, Clarification to NUREG-0578 (NRC Letter, Nov. 9, 1979), NUREG-0660, NUREG-0694, and 23
NUREG 0737.

REFERENCES

23 FSAR section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems, and subsection 12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation, have been revised to be consistent with the above.

III.D.3.4 - NUREG 0737 CONTROL ROOM HABITABILITY REQUIREMENTS

REQUIREMENT

Position

In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Clarification

1. All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the referenced Standard Review Plans (SRP) sections. The new clarification specifies that licensees that meet the criteria of the SRPs should provide the basis for their conclusion that SRP 6.4 requirements are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

2. All licensees with control rooms that meet the criteria of the following sections of the Standard Review Plan: 23

   2.2.1-2.2.2 Identification of Potential Hazards in Site Vicinity
   2.2.3 Evaluation of Potential Accidents;
   6.4 Habitability Systems,

   shall report their findings regarding the specific SRP sections as explained below. The following documents should be used for guidance:

   a. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Regulatory Power Plant Control Room During a Postulated Hazardous Chemical Release";

   b. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"; and,

   c. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

   Licensees shall submit the results of their findings as well as the basis for those findings by January 1, 1981. In providing the basis for the habitability finding, licensees may reference their past submittals. Licensees should, however, ensure that these submittals reflect the current facility design and that the information requested in Attachment 1 is provided.

3. All licensees with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, Regulatory Guides, and other references.

These licensees shall perform the necessary evaluations and identify appropriate modifications.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isolation valves (MSIV) (i.e., valve-stem leakage,
23 valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems. Other DBAs should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA.

In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

ATTACHMENT 1, INFORMATION REQUIRED FOR CONTROL-ROOM HABITABILITY EVALUATION

1. Control-room mode of operation, i.e., pressurization and filter recirculation for radiological accident isolation or chlorine release.

2. Control-room characteristics:

   a. Air volume control room.

   b. Control-room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.).

c. Control-room ventilation system schematic with normal and emergency air-flow rates.

d. Infiltration leakage rate.

e. High efficiency particulate air (HEPA) filter and charcoal adsorber efficiencies.

f. Closest distance between containment and air intake.

g. Layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions.

h. Control-room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.

i. Automatic isolation capability-damper closing time, damper leakage and area.

j. Chlorine detectors or toxic gas (local or remote).

k. Self-contained breathing apparatus availability (number).

l. Bottled air supply (hours supply).

m. Emergency food and potable water supply (how many days and how many people). 23

n. Control-room personnel capacity (normal and emergency).

o. Potassium iodide drug supply.

3. Onsite storage of chlorine and other hazardous chemicals:

   a. Total amount and size of container.

   b. Closest distance from control-room air intake.

4. Offsite manufacturing, storage, or transportation facilities of hazardous chemicals:

   a. Identify facilities within a 5-mile radius.

   b. Distance from control room.

   c. Quantity of hazardous chemicals in one container.

   d. Frequency of hazardous chemical transportation traffic (truck, rail, and barge).

5. Technical specifications (refer to standard technical specifications):

   a. Chlorine detection system.

23 b. Control-room emergency filtration system including the capability to maintain the control-room pressurization at 1/8-in. water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

RESPONSE

San Onofre Units 2&3 were designed to meet the intent of Regulatory Guide 1.78 (Rev. 0) and 1.95 (Rev. 0). Commitments to these Regulatory Guides are contained in FSAR appendix 3A. The project conformance with Standard Review Plans 2.2.1, 2.2.2, 2.2.3, and 6.4 are discussed in FSAR sections 2.2 and 6.4. These FSAR sections demonstrate the habitability of the control room considering the effect of offsite hazards. Thus, no design changes are necessary to meet the requirements of NRC position NUREG 0737, Item III.D.3.4.

The following information is provided as requested in attachment 1.

1. The control room mode of operation is pressurization and filtered recirculation for radiological accidents and isolation and filtered recirculation for toxic gas releases.

2. Control room characteristics:

   a. Volume = 293,300 ft^3 (ref. FSAR Appendix 15B)

   b. The control room emergency zone is shown on FSAR figure 6.4-1.
23 This area includes:

      (1) Control room.
      (2) Unit 2 control room cabinet area.
      (3) Unit 3 control room cabinet area.
      (4) Computer room 2.
      (5) Computer room 3.
      (6) Kitchen and operator's wash rooms.
      (7) Watch Engineer's (Shift Supervisor) office.

   c. Control room ventilation schematics are shown on figures III.D.3.4-1 and -2.

   d. Unfiltered inleakage is zero for the pressurization mode and 965 ft^3/min in the isolated mode.

e. Filter efficiencies are (%)

	Pressurizing Air	Recirc. Air
HEPA	99	99
Charcoal		
Elemental I	99*	95
Organic I	99*	95
Particulate I	99	99

*based on two 2-inch filters in series.

f. Approximately 135 ft.

g. A layout drawing of the site is shown on figure III.D.3.4-3.

h. Shielding is addressed in the response to item II.B.2.

i. The control room isolation dampers close within 6 seconds of receipt of an isolation signal. Leakage rates into and out of the control room envelope are listed in table III.D.3.4-1.

j. Redundant toxic chemical detectors are provided in the normal control room air intake to detect carbon dioxide, aqueous ammonia, butane, and chlorine. The detectors meet single failure criteria. Alarms and control logic are provided to warn the operators and automatically isolate the control room (see FSAR paragraph 6.4.3.4) when the above chemicals are present in hazardous quantities. Emergency portable breathing apparatus is also provided for the control room operators, in accordance with Regulatory Guide 1.78, Position C-13. 23

k. Protective clothing, respirators and self-contained breathing apparatus are provided for at least 9 persons which is the minimum operating shift crew size for two-unit operation.

l. Six hours of bottled air is available for the emergency control room personnel.

m. Food, water, medical supplies and sanitary facilities are provided for five persons for five days.

n. Normal control room occupancy is the minimum shift crew for two unit operation. These are:

1 Watch Engineer
1 Watch Foreman
5 Reactor Operators
2 Equipment Operators

Emergency occupancy of the control room is 5 persons.

Table III.D.3.4-1
CONTROL ROOM LEAKAGE RATES

Leak Path	Inleakage Rate At 1/8-inch WG, ft^3/min	Outleakage Rate At 1/4-inch WG, ft^3/min
Plaster Walls[a]	9.4	14.0
Duct-piping and Electrical Penetrations[b]	0.33	0.8
Dampers[a]	250.7	-164.0[c]
Elevator Shaft[a]	380.0	529.2
Doors[b]	315.0	450.0
No Airlock	10.0	10.0
Total	965.43	840

23

Notes:

a. Includes 20% safety factor
b. Includes 50% safety factor
c. The dampers on the suction side of the air handling unit have an inleakage rate of 375 ft^3/min. The dampers on the discharge side of the air handling unit have a leakage rate of 211 ft^3/min, which is recirculated through the suction side dampers. This results in a net inleakage rate of 164 ft^3/min in the control room envelope.

o. Potassium iodide is stored at various locations around the plant site to facilitate distribution in an emergency. Additional details are provided in the response to NRC question 432.50.

3. Onsite Storage of Chlorine and Other Hazardous Chemicals

a. The chemicals stored onsite, the total amount stored and size of chemical container are listed in table III.D.3.4-2. Sodium hypochlorite is used at the San Onofre site and there are no other services for gaseous chlorine onsite.

b. The distance from each chemical storage facility to the control room intake is listed in table III.D.3.4-2.

4. Offsite manufacturing, storage, or transportation facilities of hazardous chemicals.

FSAR Section 2.2 provides all of the information requested. 23

5. Technical Specifications

a. The toxic gas detection system will be included in the Technical Specifications (3/4.3.3.7).

b. The control room emergency filtration system is included in the Technical Specifications (3/4.7.5). The standard technical specifications do not currently require time response testing of the isolation dampers.

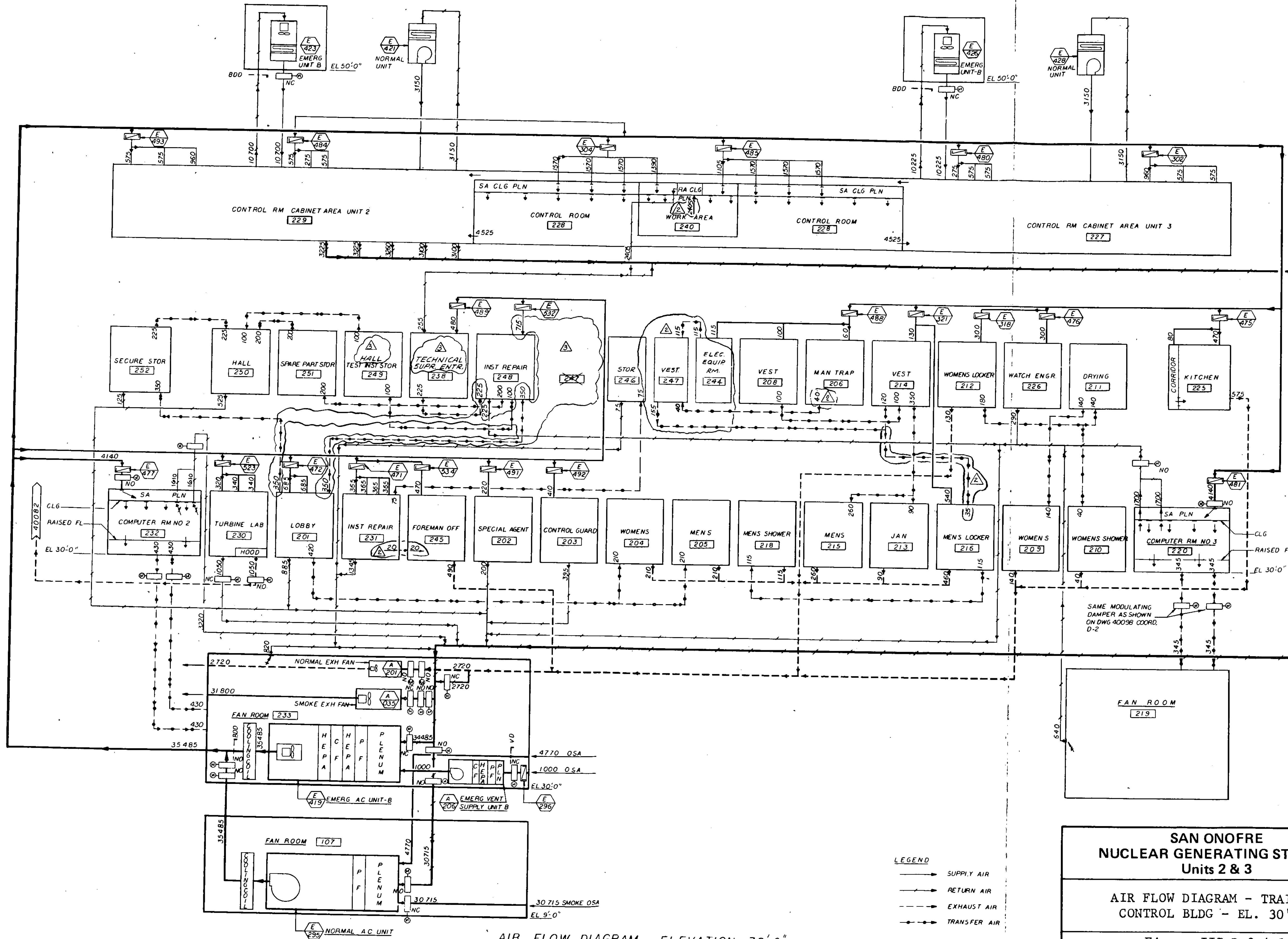
REFERENCES

FSAR sections 2.2, 6.4, and 9.4 and appendix 3A. No FSAR changes were made.

Table III.D.3.4-2
ONSITE CHEMICAL STORAGE FACILITIES AND DISTANCE
FROM CONTROL ROOM INTAKE

23

Chemical	Storage Facility	Method of Connection to System Serviced	Distance From Control Room Air Intake In Feet
Nitrogen	Compressed, liquified gas in 91.800 lb capacity tank @-320F and 245 lb/in.^{2}g	See FSAR figure 3.6-1	394 feet
Hydrogen	Compressed gas stored in 7620 scf cylinders @ 2450 lb/in.^{2}g, 70F	See FSAR figures 3.6-1 and 9.3-9	341 feet
Carbon Dioxide	Compressed, liquified gas stored in 13-ton capacity storage tank @ 300 lb/in.^{2}g, 0°F	See FSAR figure 9.5-2	112 feet
Diesel Oil	350 gal tank ambient temperature and pressure	See FSAR figure 9.5-1	495 feet
Ammonia (aqueous)	29.4% aqueous solution, 3000 gal tank, ambient temp. and press.	See FSAR figure 10.4-3	230 feet
Hydrazine (aqueous)	35% aqueous solution, 55 gal drum, ambient temp. & press.	55 gallon drums stored on ground floor of turbine bldg at el. + 7.0 ft.	72 feet
Sulfuric acid	66°Be in 10,000 gal tank, ambient temperature and pressure	See FSAR figure 9.2-2	220 feet
Halon 1301	Compressed gas stored in 140 lb capacity cylinders	See FSAR figure 9.5-2	Release inside control building

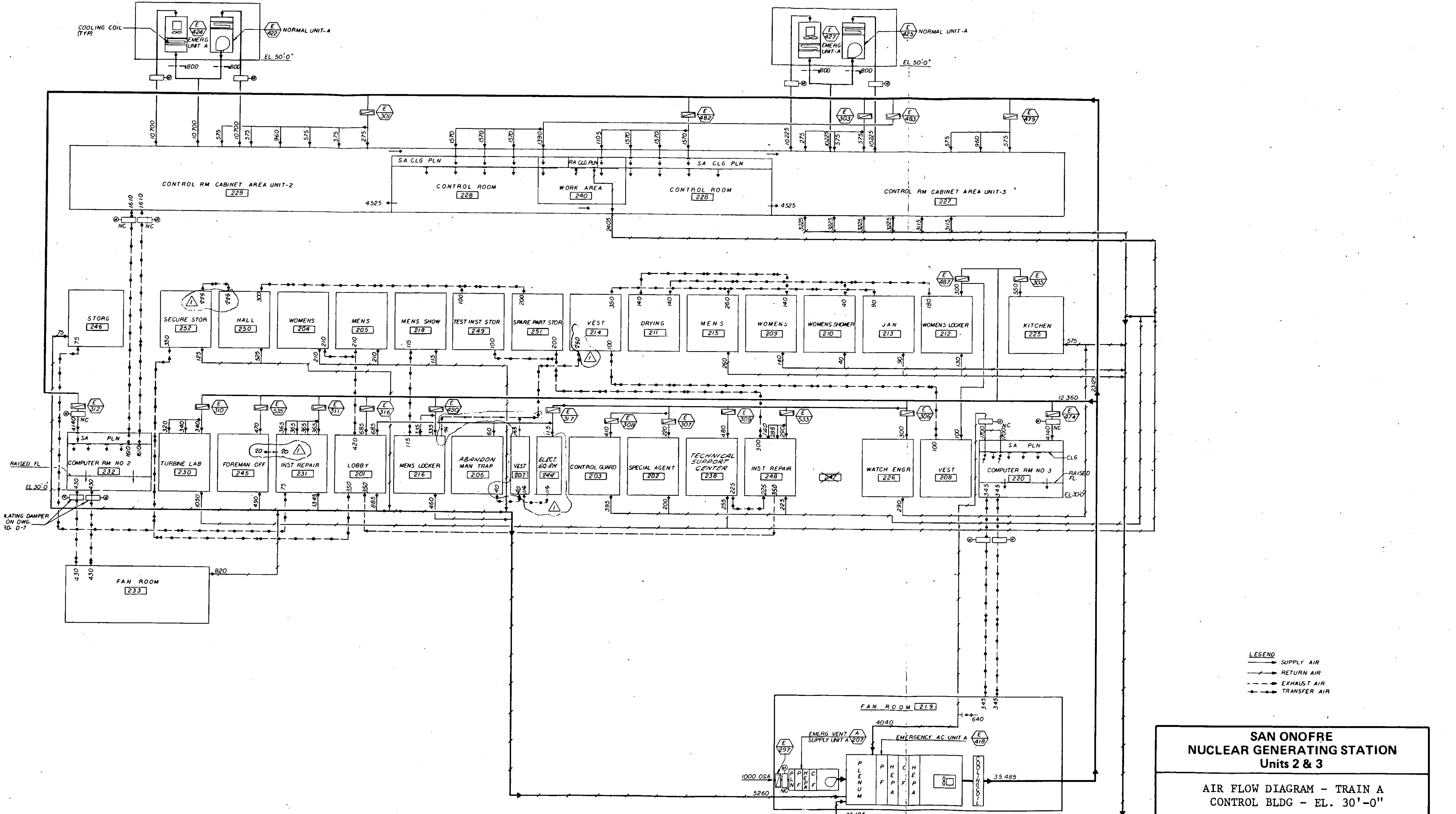


AIR FLOW DIAGRAM — ELEVATION 30'-0"

SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

AIR FLOW DIAGRAM - TRAIN B
CONTROL BLDG - EL. 30'-0"

Figure III.D.3.4-1

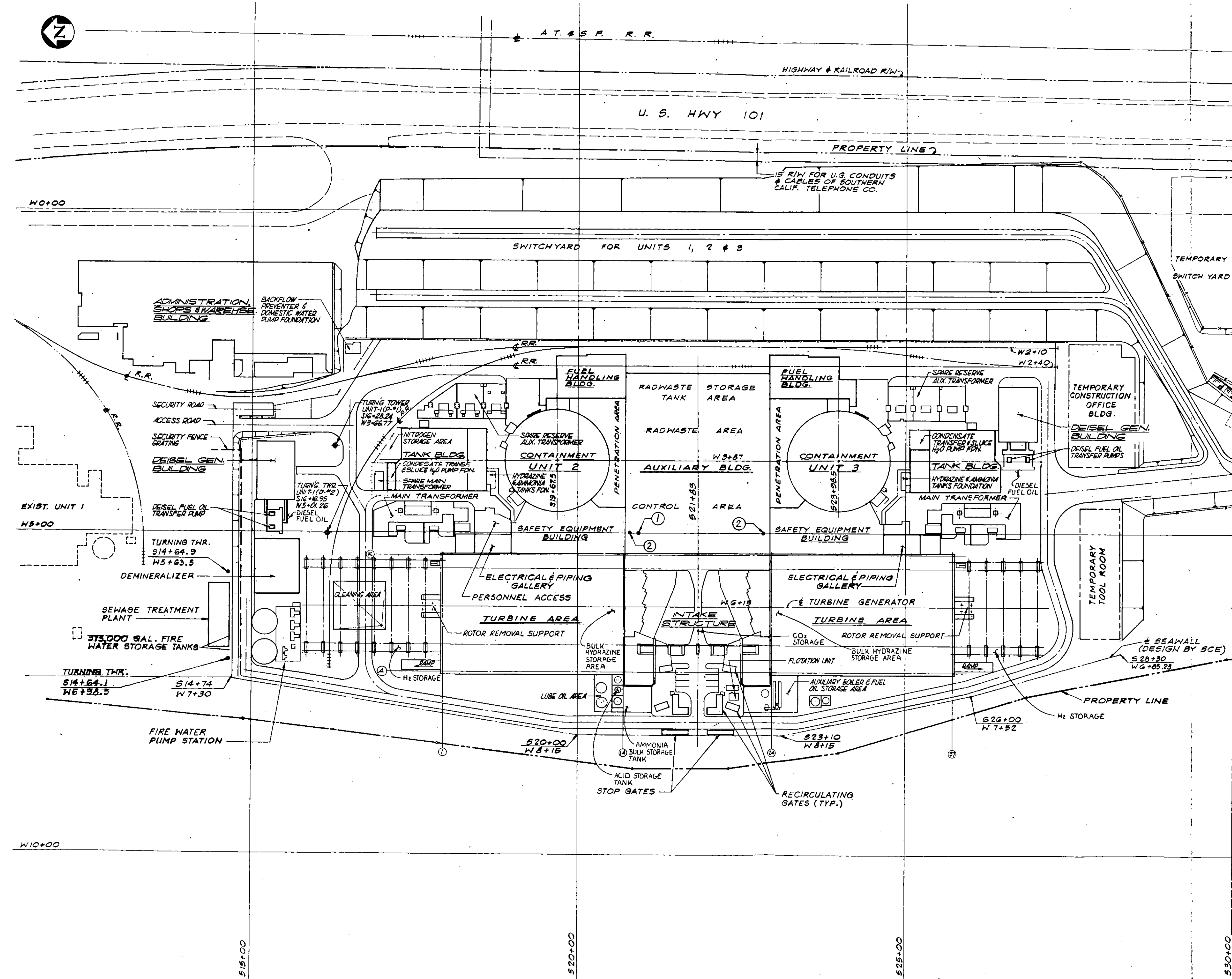


AIR FLOW DIAGRAM — ELEVATION 30'-0"

SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

AIR FLOW DIAGRAM - TRAIN A
CONTROL BLDG - EL. 30'-0"

Figure III.D.3.4-2



LEGEND:

1. CONTROL ROOM AIR INTAKE NORMAL (EL 31'-6")

2. CONTROL ROOM AIR INTAKES EMERGENCY (EL. 42'-0")

50 0 50 100

GRAPHIC SCALE IN FEET

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3

SITE LAYOUT

Figure III.D.3.4-3

FSAR

CHANGE PAGES

Table 1.7-1
NONPROPRIETARY EI&C DRAWINGS INCORPORATED
BY REFERENCE (Sheet 9 of 56)

Drawing Number	Rev. No.	Date	Title	Section Reference
30558	3	04/18/79	Elem Diag Reactor RC Loop 2 Hot Leg Injection Drain Sol	
30559	2	04/18/79	Elem Diag Reactor RC Loop 1 Hot Leg Injection Drain Sol	
30562	2	10/09/78	Elem Diag Reactor Pressurizer Press Cont (ESF)	
30563	1	10/09/78	Elem Diag Reactor Pressurizer Press Cont (ESF)	
30564	2	10/27/78	Elem Diag Reactor Pressurizer Press Cont (ESF)	
30565	1	10/12/78	Elem Diag Reactor Pressurizer Press Cont (ESF)	
30571	2	09/25/78	Elem Diag Reactor Chem Spray & Iodine Removal Sys Cont	
30572	2	06/06/78	Elem Diag Reactor Chem Spray & Iodine Removal Sys Cont	
30573	2	04/24/79	Elem Diag Reactor Safety Injection Sys Vlv Pos Ind	
30575	4	05/09/79	Elem Diag Reactor Safety Injection Sys Vlv Pos Ind	
30577	2	03/05/79	Elem Diag Reactor Pressurizer Level Cont	
30579	1	09/25/78	Elem Diag Reactor Hand Oper Valve Ind	
30580	2	02/23/79	Elem Diag reactor Aux Quench TK Makeup Iso Valve Ind	
30581	2	09/18/79	Elem Diag for Valve HV8249-2 (Unit 2)	7.3.1.1.6
30582	0	04/15/80	Elem Diag Reactor - Reactor Head and Pressurizer Vent Valves (Sh 1)	

23

Table 1.7-1
NONPROPRIETARY EI&C DRAWINGS INCORPORATED
BY REFERENCE (Sheet 9a of 56) 23

Drawing Number	Rev. No.	Date	Title	Section Reference
30582	0	04/15/80	Elem Diag Reactor - Reactor Head and Pressurizer Head Vent Valves (Sh 2)	
30583	0	04/15/80	Elem Diag Reactor - Reactor Head and Pressurizer Head Vent Valves (Sh 1)	
30583	0	04/15/80	Elem Diag Reactor - Reactor Head and Pressurizer Head Vent Valves (Sh 2)	
30584	3	04/18/79	Elem Diag for Pump P20 (Unit 2)	7.3.1.1.3
30587	2	11/07/78	Elem Diag for Valve HV9398 (Unit 2)	7.3.1.1.3

23

Table 1.7-1
NONPROPRIETARY EI&C DRAWINGS INCORPORATED
BY REFERENCE (Sheet 36 of 56) 9

Drawing Number	Rev. No.	Date	Title	Section Reference
410-6-504	2	05/03/76	Fuel Handling Bldg Pump Room Emer Cooling Units-Fan (E441, E442) (5111-0100-38)	7.3.1.1.11
410-6-509	2	07/14/76	Emer Chiller Rooms Vent System Exhaust Fan (A055, A056) (5111-0100-43)	7.3.1.1.10
410-6-510	2	07/14/76	Emer Chiller Rooms Vent Sys Supply Fan (A053, A054) (5111-0100-44)	7.3.1.1.10 7.3.1.1.11 7.3.1.1.12.4
410-6-516	2	07/14/76	Logic Diag for A173, A174 (5111-0100-50)	7.3.1.1.12.4
410-6-517	1	07/14/76	Pump Rm Emer Cooling Units (E435, E436, E437, E438, E439, E440) (5111-0100-51)	7.3.1.1.12.4
410-6-518 and 519	2 1	07/14/76 05/03/76	Emer Chilled Water Sys Loop A Chiller (E336) (5111-0100-52, 53)	7.3.1.1.10 7.3.1.1.11 7.3.1.1.12.4
410-6-520	3	10/04/76	Emer Chilled Water Loop A Pump (P162) (5111-0100-54)	7.3.1.1.10 7.3.1.1.11 7.3.1.1.12.4
410-6 521	2	07/14/76	Emer Chilled Water Sys Loop B Chiller (E335) (5111-0100-55)	7.3.1.1.10 7.3.1.1.11 7.3.1.1.12.4
410-6-522	4	03/25/77	Emer Chilled Water Loop B Pump (P160) (5111-0100-56)	7.3.1.1.10 7.3.1.1.11 7.3.1.1.12.4
504-3-341	5	01/30/80	Aux Feedwater to Stm Gen E089 (2L-188B-CD-67)	
504-3-355	5	01/30/80	Aux Feedwater to Stm Gen E088 (2L-188A-CD-81)	

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Table 1.7-1
NONPROPRIETARY EI&C DRAWINGS INCORPORATED
BY REFERENCE (Sheet 36a of 56)

Drawing Number	Rev. No.	Date	Title	Section Reference
53013	6	08/27/79	(CR-52) Steam Gen Wtr Level Control & Feedwater & Condensate	7.3.1.1.6 7.3.1.1.7 7.4.1.2
53014	4	01/03/79	(CR 54 & 64) Turb & Saltwater & Component Cooling Water Sys	7.3.1.1.4 7.3.1.1.12.1 7.3.1.1.12.2
53016	5	08/27/79	Plant Prot Sys & Reactor Coolant Pumps 2CR-56	

Table 1.7-1
NONPROPRIETARY EI&C DRAWINGS INCORPORATED
BY REFERENCE (Sheet 56 of 56)

Drawing Number	Rev. No.	Date	Title	Section Reference
944-378 thru 944-389	1	10/26/76	Intercon/D Indx+Legend (9184-8826) Sh 1 thru 12 (B-1370-416-001-B)	7.2 7.7
944-423	1	02/14/77	PPS Bypass and Block Schematic (E-1370-411-560-E)	7.2 7.3
944-471	0	12/12/77	Aux Prot Cabinet Cbl D (D-1317-414-075-D)	7.2
944-580	2	10/6/80	Intercon/D (Stm Gen Wide Range Water Level) L-1114, L-1115, L-1124, L-1125 Sh 1 (B1370-416-044 Rev. 3)	7.5
944-581	1	7/11/80	Intercon/D (Stm Gen Wide Range Water Level) L-1114, L-1115, L-1124, L-1125 Sh 2 (B1370-416-004 Rev. 2)	7.5

23

section 3.6. Environmental design is discussed in section 3.11. Testing and inspection is discussed in paragraph 6.2.5.4. Additional design bases follow:

A. The hydrogen recombiner subsystem is designed to maintain the containment hydrogen concentration below 4.0 vol %, the lower combustible limit of hydrogen in air as specified in NRC Regulatory Guide 1.7.

B. The hydrogen purge subsystem provides the installed capability for a controlled purge of the containment atmosphere in order to maintain the hydrogen concentration below 4 vol % following a LOCA. With the purge system operating, the total doses at the exclusion area boundary and low population zone outer boundary will not exceed the guideline values of 10CFR100.

C. Hydrogen mixing is provided by the containment spray system (described in paragraph 6.2.2.1), the containment emergency fan coolers (described in paragraph 6.2.2.2), the containment dome air circulators (described in paragraph 9.4.1.2), and the containment internal structure design, which permits convective mixing and prevents local entrapment of the hydrogen for as long as accident conditions require.

23 D. The redundant hydrogen monitoring subsystem is a safety related system that is designed to measure the hydrogen concentration inside the containment at two independent locations and to alert the operator in the control room of the need to activate the hydrogen recombiners or hydrogen purge system. It meets the requirements of NUREG-0737 and provides continuous indication of
23 the containment hydrogen concentration following a LOCA. This subsystem is Quality Class II, Seismic Class I and Electrical Class 1E.

E. The redundant hydrogen recombiner and hydrogen monitoring subsystem are designed so that a single failure of any component, assuming loss of offsite power, cannot impair the ability of the system to perform its designated function.

F. The redundant hydrogen recombiner and hydrogen monitoring subsystem are designed to perform their designated functions following a design basis earthquake.

6.2.5.2 System Design

6.2.5.2.1 General Description

The total system for control of combustible hydrogen concentrations in the containment following a LOCA consists of a hydrogen monitoring subsystem that measures the containment atmosphere hydrogen concentration, a hydrogen recombiner subsystem that provides the primary means of reducing containment hydrogen concentrations, and a hydrogen purge subsystem that provides the installed capability for a controlled purge of the containment atmosphere.

Table 6.2-31
COMBUSTIBLE GAS CONTROL SYSTEM DESIGN PARAMETERS (Sheet 2 of 4)

Parameter	Value
Hydrogen monitoring subsystem	
Hydrogen analyzer	
Number	2
Number required for operation	1
Measurement range, volume %	0-10%
Accuracy, ± Vol %	± 5% FS
Design pressure, $lb/in.^2g$	60
Design temperature, °F	300
Valves	
Design pressure, $lb/in.^2g$	60
Design temperature, °F	300
Piping	
Design pressure, $lb/in.^2g$	60
Design temperature, °F	300
Hydrogen purge subsystem	
Hydrogen purge supply unit	
Type	Built-up unit
Number	1
Flowrate each (standard ft^3/min)	50

23

D. An exhaust chamber that mixes and dilutes the hot effluent with containment air to lower the temperature of the discharge stream.

E. An outer enclosure to protect the unit from impingement by containment spray.

Containment atmosphere is heated within the recombiner in a vertical duct, causing it to rise by natural convection. As it rises, replacement air is drawn through intake louvers downward through a preheater section, which will temper the air and lower its relative humidity. The preheated air then flows through an orifice plate, sized to maintain a 100-standard ft^3/min flowrate, to the heater section. The airflow is heated to above 1150F, the reaction temperature for the hydrogen-oxygen reaction, and any free hydrogen present reacts with atmospheric oxygen to form water vapor. After passing through the heater section, the flow enters a mixing section, which is a louvered chamber where the hot gases are mixed and cooled with containment atmosphere before the gases are discharged directly into the containment. The air-discharge louvers are located on three sides of the recombiner. To avoid short circuiting of previously processed air, no discharge louvers are located on the intake side of the recombiner. Figure 6.2-40 shows the hydrogen control system.

Tests have verified that the hydrogen-oxygen recombination is not a catalytic surface effect associated with the heaters, but occurs due to the increased temperature of the process gases. As the phenomenon is not a catalytic effect, saturation of the unit cannot occur. Results of testing a prototype electric hydrogen recombiner are given in reference 15 and production unit test results are given in references 16, 17, and 18. There is no difference between the hydrogen recombiner unit to be installed in San Onofre Units 2 and 3 and the unit for which the qualification tests were conducted.

6.2.5.2.2.2 <u>Hydrogen Monitoring Subsystem</u>. The hydrogen monitoring subsystem for each unit consists of two completely redundant trains. Each train consists of a hydrogen sensor, an electronic subassembly and remote readout/alarms (see figure 9.3-6). The electronic subassemblies for trains A and B are mounted in a cabinet located in the control room. Physical
23 and electrical separation is provided between train A and train B components inside the electronic subassemblies cabinet.

A bottled nitrogen and hydrogen supply is used to calibrate the sensors inside the containment at those intervals specified in subsection 16.3/4.6.4. The gas bottles, required piping and respective containment isolation valves are shown on figures 9.3-6 and 9.4-2, sheet 1.

The two redundant sensors are spaced approximately 90° apart at the periphery of the containment, near the respective calibration line containment isolation valves, in the vicinity of penetrations 16A and 23B (see figure 6.2-43).

An absolute pressure transmitter, for each train, is mounted with the respective sensor and provides a pressure compensation signal to the electronic subassembly. 23

Operation of the hydrogen gas analyzer is based on the chemical reaction that takes place in an electrochemical cell when exposed to hydrogen containing atmosphere. The electrochemical cell is essentially a hydrogen-fueled battery consisting of two electrodes in an acid electrolyte. A membrane covering the sensing electrode meters the hydrogen flow to the electrode at a rate that is proportional to the hydrogen partial pressure in the ambient atmosphere. Hydrogen is ionized on the sensing electrode. The counter electrode is a metal oxide electrode that reacts with the hydrogen ion to plate out the metal. The range and accuracy of the hydrogen analyzer is given in table 6.2-31. Failure modes and effects analysis are given in table 6.2-32.

6.2.5.2.2.3 Hydrogen Purge Subsystem. The hydrogen purge exhaust train consists of an upstream and downstream HEPA filter, charcoal adsorption filter, fan, electric motor drive, electric exhaust fan heater, and associated piping, valves, ductwork, dampers, instruments, and controls. The hydrogen purge supply train consists of a supply fan, electric motor, drive, a prefilter, associated piping, valves, ductwork, dampers, instruments, and controls. The isolation valves are the only moving parts located inside the containment. The hydrogen purge subsystem is designed to bleed and feed containment atmosphere at a rate of 50 standard ft^3/min.

The hydrogen purge supply and exhaust lines are located in missile-protected areas, and are designed to circulate air in a manner that prevents either containment spray or sump water from entering the ducts.

Should it be necessary to use the hydrogen purge subsystem, operational considerations and site meteorology would determine the timing and duration of purges. In any case, sufficient purging would be performed to maintain the hydrogen concentration in the containment atmosphere below 4 volume percent.

The hydrogen purge subsystem was designed and constructed to Seismic Category II requirements except for containment penetrations which are Seismic Category I. Failure modes and effects analysis is not applicable to this subsystem.

6.2.5.2.3 System Operation

6.2.5.2.3.1 Normal Operation. Except for testing, the system normally is not operated.

6.2.5.2.3.2 Accident Operation. Following a LOCA, the containment emergency fan coolers, the containment dome air circulators, and the containment spray system are automatically started. These systems serve to minimize localized hydrogen buildup within the containment as well as
23 remove fission products and reduce containment pressure. Within 30 minutes after a LOCA both redundant hydrogen analyzers are activated to monitor hydrogen levels and to alert the operator in the control room when operation of the hydrogen recombiners or hydrogen purge system is required; i.e., at or below a hydrogen concentration of 4 vol.%.

The recombiners are manually started before hydrogen concentration reaches 2.0 vol %. Figure 6.2-44 shows that this occurs at approximately 11 days after the LOCA. In the unlikely event that both recombiners fail, the
23 hydrogen purge subsystem can be manually put into operation.

6.2.5.3 Design Evaluation

Safety evaluations are listed to correspond to design bases.

A. Two completely separate and redundant recombiner subsystems are provided, each powered from a separate Class IE electrical bus. Thus, a single failure will not prevent the recombiners from performing their safety function as shown in table 6.2-32.

23 Although the recombiners are to be started 11 days after a LOCA, inadvertent actuation immediately after a LOCA will not damage the recombiners in any manner, nor will their capability to perform their design function be hindered or impaired. Figure 6.2-44 shows the hydrogen concentration vs. time within the containment
23 and the effect of one or two recombiners starting as late as 14 days following a LOCA and operating at the design flowrate of 100 ft^3/min per unit.

Tests have verified that recombination is not a catalytic surface effect, but that it occurs due to the increased temperature of the process gases. Poisoning of the unit by fission products or containment spray will not occur. The heater-recombiner section

Table 6.2-34
SUMMARY OF ASSUMPTIONS USED FOR HYDROGEN GENERATION FROM RADIOLYSIS

1. Reactor power level is 3560 MWt.

2. An insignificant quantity of hydrogen is generated due to the radiolysis from the noble gas isotopes.

3. The guidelines as set forth in Regulatory Guide 1.7 were followed:

   a. 100% of the noble gases are released to the atmosphere

   b. 50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water

   c. $G(H_2)$ is 0.5 molecules/100 eV

   d. The following percentage of fission product radiation energy is absorbed by the coolant:

Percentage	Radiation Type	Location of Source
0	Beta	Fuel rods
100	Beta	Coolant
10	Gamma	Fuel rods
100	Gamma	Coolant

Zinc in the containment is in two forms: zinc base paint and galvanized steel. Use of galvanized steel and zinc rich primers in containment is limited as much as practical. Maximum use of Epoxy paint is made as a substitute for zinc rich primers. During the NaOH addition phase, the containment is sprayed with a borated solution adjusted to a pH range of 9-10 with sodium hydroxide. During the long term recirculation phase, the pH of the spray will drop to the range of 8-9. The hydrogen generation rates from zinc base paint and galvanized steel in this environment are given in table 6.2-35. The hydrogen generation rate from corrosion of zinc is shown in
23 figure 6.2-46.

Use of aluminum inside the containment has been kept to an absolute minimum. The hydrogen generation rate from corrosion
23 of aluminum is shown in figure 6.2-46.

The lines penetrating containment are provided with power-operated isolation valves inside containment (and manually operated isolation valves outside containment) to allow remote isolation from the control room. These valves are locked closed during normal operation. System startup requires both manual and remote valve positioning. A complete discussion of the isolation valve provisions is presented in subsection 6.2.4.

A differential pressure gauge is provided across the vent filters to allow detection of filter clogging. Local temperature and pressure indicators are provided in the exhaust line to aid in the operation of the system.

6.2.5.5.3 Hydrogen Monitoring Subsystem

Two redundant hydrogen analyzers are provided to monitor the containment atmosphere following a design basis event to serve as a basis for actuating the hydrogen recombiners or hydrogen purge system. Hydrogen concentration is displayed both at the cabinet and in the control room. High hydrogen concentration and system malfunction alarms are provided in the control room and are indicated by lights at the cabinet. Both channels can be initiated from either the control room or the cabinet. 23

6.2.6 CONTAINMENT LEAKAGE TESTING

6.2.6.1 Containment Integrated Leakage Rate Test

After construction of the containment is completed and all electrical and piping penetrations, the equipment hatch, and personnel locks are in place, the structural integrity test and the preoperational integrated leakage rate test (ILRT or Type A test) are conducted. The structural integrity test is discussed separately in paragraph 3.8.1.7. A general inspection of the accessible interior and exterior surfaces of the containment structures and components is performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leaktightness. If there is evidence of structural deterioration, Type A tests are not performed until corrective action is taken in accordance with repair procedures, nondestructive examinations, and tests as specified in the applicable code specified in 10CFR50.55a. The maximum acceptable containment leakage rate resulting from Type A testing is specified in paragraph 16.3.6.1.2.

The containment is equipped with penetrations for containment pressurization pressure sensors, and verification test piping to be used during the integrated leakage rate test (ILRT). These penetrations are locally leakage rate tested as part of the penetration testing schedule (see paragraph 6.2.6.2). The ILRT instrumentation and pressurization equipment is not permanently installed. Instrumentation can be brought in, calibrated, and installed, whenever the containment interior is accessible; all piping, wiring, and instrument mounts are permanently installed. Pressurization equipment is mounted outside containment on a pad adjacent to the containment wall.

Table 7.5-1
SAFETY-RELATED DISPLAY INSTRUMENTATION (Sheet 16 of 32)

System	Parameter Measured(b)	No. of Channels	Range(c)	Accuracy(d) (%)	Type of Readout(e)	Location(f)	Panel No.	Power 1E Bus	RPS(h)	ESF(h)	ASI(h)	PPDI(h)	PAMI(h)	BISI(h)(i)
	Containment emergency cooling unit on/off	4	-	-	L	Control room	CR60	Yes		X				X
	CCW from emergency cooling units	4	-	-	L	Control room	CR60	Yes		X				
	CCW from emergency cooling units	4	-	-	L	Control room	CR57	Yes		X				X
	CCW to emergency cooling units	4	-	-	L	Control room	CR60	Yes		X				
	CCW to emergency cooling units	4	-	-	L	Control room	CR57	Yes		X				X
	Emergency radiation monitoring system	3	1 to 10^6 mrem/h	30	I	Control room	L-103	Yes					X	
	Emergency radiation monitoring system	1	1 to 10^6 mrem/h	30	R	Control room	L-103	Yes					X	
	Containment high range radiation monitors	2	1 to 10^8 Rads/h	20	I	Control room	L-405	Yes					X	
	Containment high range radiation monitors	1	1 to 10^8 Rads/h	20	R	Control room	L-405	Yes					X	
	Containment Pressure	2	0-200 lb/in^2g	5	I	Control room	CR57	Yes					X	
	Containment Pressure	1	0-200 lb/in^2g	5	R	Control room	CR59	Yes					X	
	Containment Normal sump	2	10'-2" to 16'-7"	±2	I	Control room	CR57	Yes					X	

2/81

7.5-19

AMENDMENT 2

Table 7.5-1
SAFETY-RELATED DISPLAY INSTRUMENTATION (Sheet 16A of 32)

System	Parameter Measured(b)	No. of Channels	Range(c)	Accuracy(d) (%)	Type of Readout(e)	Location(f)	Panel No.	Power 1E Bus	RPS(h)	ESF(h)	ASI(h)	PPDI(h)	PAMI(h)	BISI(h)(i)
	Containment Normal Sump	1	10'-2" to 16'-7"	±2	R	Control Room	CR59	Yes					X	
	Containment Emergency Sump	2	1'-6" to 20'-4"	±2	I	Control room	CR57	Yes					X	
	Containment Sump	1	11'-6" to 20'-4"	±2	R	Control room	CR59	Yes					X	
	Containment Area Water Level	2	17'-8" to 29'-8"	±2	I	Control room	CR57	Yes					X	
	Containment Area Water Level	1	17'-8" to 29'-8"	±2	R	Control room	CR59	Yes					X	
	Containment Hydrogen	2	0-10%	±5	I	Control room	CR57	Yes					X	

Table 7.5-2
POST-ACCIDENT MONITORING PARAMETERS MONITORED (Sheet 1 of 3)

Parameter	Type of Readout	Number of Channels	Range	Indicator Accuracy			Location
				Normal (%)	Seismic (%)	DBA (%)	
Pressurizer pressure	Indicator Recorder	2 1	0-3000 lb/in.^{2}a 0-3000 lb/in.^{2}a	±2 ±2	±2 ±2	±18 ±18	Control room Control room
Pressurizer level	Indicator Recorder	2 1	0-100% 0-100%	±2 ±2	±4 ±4	±3 ±3	Control room Control room
Steam generator pressure	Indicator Recorder	4 2	0-1200 lb/in.^{2}g 0-1200 lb/in.^{2}g	±2 ±2	±7 ±7	±9 ±9	Control room Control room
Steam generator level	Indicator Recorder	4 2	0-100% 0-100%	±2 ±2	±3 ±3	±3 ±3	Control room Control room
Spray system temperature	Indicator Recorder	2 1 (a)	0-400F 0-400F	±2 ±2	±2 ±2	±2 ±2	Control room Control room
Spray system pressure	Indicator Recorder	2 1 (a)	0-550 lb/in.^{2}g 0-550 lb/in.^{2}g	±2 ±2	±7 ±7	±2 ±2	Control room Control room
Low pressure safety injection header temp	Indicator Recorder	2 1	0-400F 0-400F	±2 ±2	±2 ±2	±2 ±2	Control room Control

a. One recorder with two pens, one for temperature and one for pressure.
b. One recorder with two pens, one for pressure and one for tank level.
c. Each channel monitors airborne particulate, iodine, and gaseous activity.
d. Operable spare sensor.

23

Table 7.5-2
POST-ACCIDENT MONITORING PARAMETERS MONITORED (Sheet 2 of 3) 23

Parameter	Type of Readout	Number of Channels	Range	Indicator Accuracy Normal (%)	Indicator Accuracy Seismic (%)	Indicator Accuracy DBA (%)	Location
Reactor coolant temperature (T-cold)	Indicator	4	465-615F	±2	±2	±2	Control room
	Recorder	2	465-615F	±2	±2	±2	Control room
Containment pressure	Indicator	2	-4 to 85 lb/in.^{2}g	±2	±7	±2	Control room
	Recorder	1 (b)	-4 to 85 lb/in.^{2}g	±2	±7	±2	Control room
Containment pressure	Indicator	2	0 to 200 lb/in.^{2}g	±2	±2	±2	Control room
	Recorder	1	0 to 200 lb/in.^{2}g	±2	±2	±2	Control room
Refueling water tank level	Indicator	2	0-100%	±2	±4	±2	Control room
	Recorder	1 (b)	0-100%	±2	±4	±2	Control room
Auxiliary feed - water flow	Indicator	2	0-800 gal/min	±2	±2	±2	Control room
	Recorder	1	0-800 gal/min	±2	±2	±2	Control room
Containment temperature	Indicator	2	0-400F	±2	±2	±2	Control room
	Recorder	1	0-400F	±2	±2	±2	Control room
Emergency radiation monitoring system	Indicator	3	1-10^6 mrem/h	±2	±2	±2	Control room
	Recorder	1	1-10^6 mrem/h	±2	±2	±2	Control room
Containment airborne radiation monitors	Indicator	2 (c)	Refer to table 7.5-1	±2	±2	±2	Control room
	Recorder	1 (c)		±2	±2	±2	Control room
Containment high range radiation monitors	Indicator	2	10° - 10^8 rads/h	±3	±3	±3	Control room
	Recorder	1	10° - 10^8 rads/h	±3	±3	±3	Control room

23

Table 7.5-2
POST-ACCIDENT MONITORING PARAMETERS MONITORED (Sheet 3 of 3) 23

Parameter	Type of Readout	Number of Channels	Range	Indicator Accuracy			Location
				Normal (%)	Seismic (%)	DBA (%)	
Subcooling margin	Indicator	2	0-200F margin or 0-3200 psia margin	-	-	-	Control room
Steam generator level, wide range	Indicator Recorder	2 1	Tube sheet to upper nozzle	±2 ±2	±3 ±3	±3 ±3	Control room Control room
Reactor coolant temperature (T-hot wide range)	Indicator Recorder	2 1	0-710F 0-710F	±2 ±2	±2 ±2	±2 ±2	Control room Control room
Containment water level	Indicator Recorder	2 1	10'2" - 16'7" 10'2" - 16'7"	±2 ±2	NA NA	±2 ±2	Control room Control room
	Indicator Recorder	2 1	11'6" - 20'4" 11'6" - 20'4"	±2 ±2	NA NA	±2 ±2	Control room Control room
	Indicator Recorder	2 1	17'8" - 29'8" 17'8" - 29'8"	±2 ±2	NA NA	±2 ±2	Control room Control room
Containment hydrogen monitor	Indicator	2	0 - 10% H_2	±5	±5	±5	Control room

23

9.3.6 POST-ACCIDENT SAMPLING SYSTEM

9.3.6.1 Design Basis

The post-accident sampling system design bases are as follows:

A. The post-accident sampling system (PASS) provides a means to obtain and analyze pressurized and unpressurized reactor coolant liquid and containment atmosphere samples within 3 hours after a decision is made to take a sample. A reactor coolant sample can be drawn directly from the reactor coolant system (RCS) whenever the RCS pressure is between 200 lb/in.^{2}g. and 2485 lb/in.^{2}g. At pressures below 200 lb/in.^{2}g., RCS samples can be drawn via the ESF pump miniflow recirculation line sample line. A containment atmosphere sample can be drawn with containment pressure between 10 lb/in.^{2}a. and 75 lb/in.^{2}a.

B. The PASS provides a means to quantify the parameters identified in items 1 through 3, below, while conforming to criteria identified in items 4 through 8, below.

1. Certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines, cesiums, and non-volatile isotopes).

2. Dissolved gases (i.e., H2, O2), boron concentration, chloride
23 concentration, and pH of the reactor coolant. Total gas concentrations of up to approximately 2000 cc/kg (STP) can be measured.

3. Hydrogen levels in the containment atmosphere in the range of 0 to 10 volume %.

4. The PASS allows for sampling post-accident with resulting personnel radiation exposure not exceeding five and 75 Rem to the whole body and extremities, respectively.

5. The PASS is capable of accommodating an initial reactor coolant radiochemistry spectrum corresponding to Regulatory Guide 1.4, Revision 2, or 1.3 Revison 1 release.

6. All sample flow is returned to the containment to preclude unnecessary contamination of other auxiliary systems and to ensure that high level waste remains isolated within the containment.

7. Outside of the containment isolation valves and the safety injection system isolation valve to the sampling system, components and piping are designed to Quality Group D non-seismic requirements. This complies with NUREG-0578 Section 2.1.8.a for equipment downstream of the second isolation valve from safety code systems.

8. Process Sampling points capability and design data for the PASS are shown in table 9.3-14. Table 9.3-15 provides the analysis capabilities for the PASS.

Table 9.3-14

PROCESS SAMPLING POINTS, CAPABILITY, AND DESIGN DATA

Sample Origin	Type of Sample Cooler Provided	Discrete Sample Capability(a)	Pressurized Sample Capability	Continuous Online Analyses Provided	Mode of Sample Removal	Nominal Temperature (°F)	Nominal Pressure ($lb/in.^2g$)	Figure Number
Primary Sampling System	Shell and Tube	Yes,(e)	Yes	No	(b)	120-621	2235	9.3-11
Containment Atmosphere	None	Yes,(e)	Yes	No	(b)	60-290	-5 to 60	9.3-11

23

Notes:

a. Letter refers to the discrete samples listed in table 9.3-15.

b. Septum plugs are provided in the diluted sample vessels. These plugs allow sample withdrawal using a syringe.

c. Undiluted, cooled and depressurized liquid grab sample to be provided for chloride analysis and backup to inline analytical capability.

d. Undiluted, depressurized containment atmosphere grab sample to be provided for backup to inline analytical capability.

e. See Table 9.3-15.

23

Table 9.3-15

DISCRETE ANALYSIS CAPABILITIES

A. Liquid Analyses

1. pH
2. Boric Acid Concentration
3. Total Dissolved Gas Concentration
4. Fission Product Content and Activity
5. Dissolved Hydrogen
6. Dissolved Oxygen
9. Chloride [a]

B. Containment Atmosphere

7. Hydrogen
8. Fission Product Content and Activity

21

C. Sample lines connected to ASME Section III code class lines are constructed in accordance with the same code class up to and including the first normally closed, automatic, or throttling valve.

D. The pressure and temperature ratings of the sample lines and components correspond to the rating of the particular system being sampled up to and including the system throttling valves. Downstream of the throttling valve or orifice, sample piping pressure and temperature are reduced to correspond to downstream process conditions, and relief protection is provided where appropriate.

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23 (a) By undiluted grab sample within 4 days.

9.3.6.2 System Description

9.3.6.2.1 General Description

The post-accident sampling system is illustrated schematically by figures 9.3-10 and 9.3-11.

The post accident sampling system will be located at the 24-foot elevation in a separate room in the radwaste building located between Units 2&3. Figure 9.3-17 shows a sketch of the room arrangement which locates the sample station and isolation valves in a shielded room with control panels on the safe side of the biological shielding.

Location of sample points are shown on figure 9.3-10. The post-accident sampling system consists of a sample station, pumps, sample vessels, heat exchanger, burette, strainer, germanium crystal, and multichannel analyzer.

9.3.6.2.2 Component Description

9.3.6.2.2.1 Sample Station. The sample station is a free standing, totally enclosed metal panel measuring 6 ft long by 6 ft high by 4 ft deep. The enclosure contains system tubing, valves, components, and instrumentation necessary to provide chemistry analysis capability for reactor coolant and containment atmosphere sampling per NUREG-0578, Section 2.1.8.a. Louvers are provided in the cabinet and sized to pass up to 333 standard ft^3/min of airflow from the surrounding room to the ventilation system exhaust connection in the upper portion of the enclosure. This airflow precludes any possible buildup of radioactive gas or H_2 gas and provides for removal of heat generated by internal components. The station is skid-mounted and is provided with removable panels or doors on all four sides to ensure easy accessibility for any necessary maintenance on system components. The station is bolted to the floor of the sample room. Radiochemistry analysis of liquid and gas samples is provided by separate germanium detectors and a single multichannel analyzer (see paragraphs 9.3.6.2.3.j and 9.3.6.2.3.k).

9.3.6.2.2.2 Sample Circulation Pump. The sample circulation pump is a peristaltic type positive displacement pump. This pump is capable of pumping liquids and/or gases. The pump will be used in the total gas, hydrogen, and oxygen gas analyses operations to strip the gases out of solution in the sample fluid and circulate them through the hydrogen and oxygen analyzers.

9.3.6.2.2.3 Surge Vessel Pump. The surge vessel pump is a progressing cavity (helical) pump. The pump is used to pump down the surge vessel contents to the reactor coolant drain tank and is also used in the calibration operation of the pH meter in the liquid sample line.

9.3.6.2.2.4 Containment Sample Pump. The containment sample pump is a vacuum pump/compressor unit that operates as a positive displacement

9.3.6.2.2.11 Strainer. The strainer is designed to remove insoluble particles which may cause sample station chemistry instrumentation to become plugged. The strainer can be backflushed with demineralized water remotely by operation of valves at the control panel.

9.3.6.2.3 Instrument and Control Description

The system is designed to be controlled remotely from a designated post-accident sample control area where the control panels are located. The control panels are free-standing, straight front, vertical, metal panels measuring 7 ft high by 2-1/2 ft deep with a total length of about 17 feet. All sample system non-code isolation valves, other valves outside containment, and pumps are controlled from these panels. 23 Indication of all process parameters and chemistry readouts are displayed on the panels. To facilitate system operability all controls and indications are arranged in a mimic of the system. Positions of code isolation valves and other valves within the containment, which are controlled in the main control room, are indicated on the mimic panel. The control panels are bolted to the floor in the control area. The following is a description of the instrumentation and controls for the system.

A. The depressurized liquid sample vessel, surge vessel, and stainless steel burette are equipped with level indication instrumentation to monitor for total gas, dilution, flushing, and calibration operations.

B. The sample vessel/heat exchanger tube side outlet is equipped with a temperature measuring device to indicate adequate sample cooling for downstream instrumentation protection.

C. All of the containment atmosphere sample piping is heat traced to limit plateout of radioisotopes which would result from condensation of containment atmosphere vapor. (Heat tracing is controlled at the control panels.)

D. The containment sample pump discharge line and the sample vessel/heat exchanger outlet line are equipped with a flow measuring device to monitor proper sample purging flow rate.

E. All PASS pumps are equipped with handswitches at the control panels.

F. The boronmeter is a specific gravity measuring device which determines and indicates the amount of boron (ppm) present in the liquid sample fluid.

G. The pH meter determines and indicates pH in the liquid sample fluid.

H. The H_2 analyzers are thermal conductivity devices that determine and indicate the volume percent of H_2 present in the containment atmosphere and liquid sample.

I. The O_2 analyzer is a paramagnetic device that determines and indicates the volume percent of O_2 present in the sample fluid gas removed from the liquid sample.

J. The germanium crystal packages utilize hyperpure crystals that monitor the reactor coolant and containment atmosphere sample's gamma spectrum. The crystals are used to analyze the presence of certain isotopes (e.g. Kr-85, I-131, I-133, Xe-133, Cs-137, Cs-134, Ba-140, and La-140) helpful in identifying a failed fuel accident condition.

K. The multichannel analyzer interprets the signals generated by the germanium crystals (j) and prints a spectrum analysis of the liquid and containment atmosphere samples for certain pre-programmed isotopes. Outputs from the multichannel analyzer will also be fed to the health physics computer system.

L. All pneumatically and solenoid operated valves have hand switches at the control panel.

M. Control switches are provided to automatically isolate the high pressure reactor coolant inlet to prevent system overpressurization.

23 N. To protect the normal sampling station from the high level source terms present immediately following an accident (which may occur while normal sampling is in progress) each stop valve on the liquid sample line going to the normal sample station is shut automatically by dual range area radiation monitors viewing these lines where they are routed through the PASS room.

9.3.6.2.4 System Operation

9.3.6.2.4.1 Operation During Plant Accident Conditions. Operator actions to collect and analyze reactor coolant and containment atmosphere samples during post-accident conditions consist of (1) remote operation of valves and components, (2) inline measurement of chemistry and radiochemistry parameters, and (3) preparation and handling of diluted samples for onsite radiochemistry analyses. Operational considerations for reactor coolant and containment atmosphere sampling are given below.

9.3.6.2.4.2 Reactor Coolant Sampling. Sampling of the RCS is initiated by opening system isolation valves (including containment isolation valves using CIAS override, if necessary) and purging a reactor coolant through a sample vessel/heat exchanger, where it is cooled, through a throttle valve to reduce the pressure, through the inline chemistry and radio-chemistry analysis equipment to the reactor coolant drain tank. At reactor coolant pressures (less than 200 lb/in.^{2}g.) containment sump sample flow is purged in the same manner using the HPSI pump discharge connection. 23
A float valve downstream of the throttle valve allows for automatic venting of gases coming out of solution. This venting is required in order to prevent gas bubble interference with flow rate and chemistry measurements in the downstream instrumentation. After sufficient purging, boron concentration and pH are measured using inline instrumentation. An online hyperpure germanium crystal package is used to generate a radioisotope spectrum which is transmitted to the PASS multi-channel analyzer where 23
selected isotopic concentrations are recorded. A pressurized sample is then collected by isolating the sample vessel/heat exchanger. Total dissolved gas concentration is determined by degassing the sample. This is accomplished by depressurization and circulation by alternate operations

of the burette isolation valve and the sample circulation pump. The resulting displacement of liquid into the burette is used to calculate the dissolved gas concentration. The collected gases, which have been stripped from the liquid, are then directed through a float valve for moisture separation and circulated through hydrogen and oxygen analyzers. After recording the hydrogen and oxygen gas concentrations, the gas sample vessel which contains nitrogen, may be placed on line to dilute the gas volume. This dilution operation reduces the radiation levels such that local samples can be drawn from the gas sample vessel, if desired by injection of a syringe through a septum plug mounted in the vessel. This sample may then be transferred to the site laboratory for subsequent radioisotope quantification. Prior to sample withdrawal, additional dilution, which may be necessary for this quantification, may be performed by further nitrogen addition, circulation and venting. A small fixed volume of depressurized liquid sample (collected in four-way valve PS-216) may then be drained to the depressurized liquid sample vessel. This volume is diluted by adding demineralized water to the vessel. The sample may be withdrawn from the depressurized liquid sample vessel in the same manner as described above. The radiochemistry and gaseous measurement portions of the system are flushed with demineralized water and purged with N_2 to reduce personnel exposure during withdrawal of the gaseous and liquid samples and to reduce contamination plateout between samples.

An undiluted depressurized and cooled grab sample can be obtained downstream of the PASS station by diverting the liquid sample outlet flow through a shielded sample container located outside the South shield wall between the germanium detector units. The shielded sample container can be disconnected from the sample system and lifted to the floor above the PASS room for transfer to the station hot laboratory or to an offsite hot laboratory for chloride analysis or other confirmatory analyses. 23

9.3.6.2.4.3 Containment Atmosphere Sampling. Sampling of the containment atmosphere is initiated by opening the containment isolation valves (by overriding CIAS as necessary) and by using the containment sample pump to purge the air sample through the system obtaining H_2 concentration and online isotopic analysis. Purge flow is directed back to the containment. 23 An online hyperpure germanium crystal package, similar to that used in the reactor coolant sampling, is used to generate a readout of isotopic concentrations. If desired, a sample may be manually withdrawn from the containment air lines by first isolating a known sample volume and diluting the sample by pumping it through the containment sample vessel containing nitrogen. The initial nitrogen volume dilutes the sample to levels acceptable for withdrawal. A containment air sample may then be withdrawn from the containment sample vessel in the same manner as described previously for the reactor coolant samples.

An undiluted grab sample of containment air can be obtained downstream of the PASS station in a manner similar to that used for the undiluted liquid sample described in paragraph 9.3.6.2.4.2 above. A separate, removable shielded gas sample container will be located outside the South shield wall adjacent to the liquid grab sample cask. The shielded gas grab sample will 23

be removed from the PASS room via the access hatch for confirmatory anal-
23 yses as required either at the station hot laboratory or at a suitable off-
site hot laboratory.

9.3.6.3 Design Evaluation

Design evaluations are:

The locations of the post-accident reactor coolant and containment atmosphere sampling system are in an area of relatively low post-accident background radiation. This ensures compliance with the personnel exposure limits of NUREG-0578 during sampling and analysis. Additional plant shielding along with selective routing of interconnecting piping to the existing sampling system ensures that (1) the exposure limits for personnel are not exceeded and (2) the onsite radiochemistry analysis equipment is available for post-accident sample analyses. The sample station is also physically separated from safety-related equipment such that failure of the associated non-seismic equipment does not cause damage to the safety related equipment.

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Table 9.3.16

POST ACCIDENT SAMPLING SYSTEM (PASS)
FAILURE MODE AND EFFECTS ANALYSIS (Sheet 3 of 38)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
7	Sample circulation pump discharge isolation valve; PS-218	a) Fails open	**Mechanical binding**	Unable to isolate sample circulation pump during depressurization operation resulting in possible damage to the pump discharge tubing. Loss of ability to determine total gas, hydrogen, and oxygen concentrations in the sample fluid.	Operator	None	See 2) - b)
		b) Fails closed	Mechanical binding or pneumatic failure	Unable to circulate depressurized sample for refined total gas measurements and unable to measure hydrogen and oxygen concentrations in the sample fluid.	Power indication at the control panel if failure due to loss of power.	None	See 2) - b)
8	Burette isolation valve; PS-205	a) Fails open	Mechanical binding	Unable to control sample depressurization for total gas measurement. Unable to isolate burette for future gas measurements.	Change in burette level when burette is isolated.	None	See 2) - b)
		b) Fails closed	Mechanical binding. Loss of air or power.	Unable to obtain gas, hydrogen, and oxygen measurements.	Power indication at the control panel if failure due to loss of power.	None	See 2) - b)
9	Isolation valve; PS-219	a) Fails open	Mechanical binding	Unable to isolate the normal fixed volume of pressurized liquid sample for dissolved gas measurements.	Operator	Redundant isolation valves for all adjoining low pressure system lines.	PS-205 isolates the high pressure reactor coolant sample in lieu of PS-219. Some slight error will result in the H_2, O_2, and total gas measurements.
		b) Fails closed	Mechanical binding. Loss of air or power.	Unable to perform pressurization operation for total gas, H_2 and O_2 measurements.	Power indication at the control panel if failure due to loss of power.	None	See 2) - b)

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Table 9.3.16

POST ACCIDENT SAMPLING SYSTEM (PASS)
FAILURE MODE AND EFFECTS ANALYSIS (Sheet 4 of 38)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
10	Burette level instrument isolation valves; PS-121 PS-122	a) Fails open	Mechanical binding	No impact on normal operations. Loss of isolation of L-502 for maintenance.	Operator	None req'd	-
		b) Fails closed	Mechanical binding	Unable to measure change in burette level necessary to quantify total gas concentration in the sample fluid.	Level indication does not change during depressurization operation at the control panel.	None	Normally open. See 2) - b)
11	Burette level instrument isolation valve; PS-131	a) Fails open	Mechanical binding	See 10) - b)	See 10) - b)	None	Normally closed - See 2) - b)
		b) Fails closed	Mechanical binding	See 10) - a)	See 10) - a)	None req'd	-
12	Burette vent line pressure instrument isolation valve; PS-103	a) Fails open	Mechanical binding	No impact on normal system operations. Loss of isolation of P-505 for maintenance.	Operator	None req'd	
		b) Fails closed	Mechanical binding	Unable to monitor burette and surge vessel pressure.	No indication of burette pressure or surge vessel pressure.	None	No effect on normal system operation or analyses.
13	Hydrogen/oxygen gas line float valve; PS-126	a) Fails open	Valve internal damage	Possible moisture damage to in-line H_2 meter and O_2 meter.	Possible erratic H_2 and O_2 readings.	None	If water reaches the H_2 or O_2 meters the instruments would no longer function properly. See 5) - a)
		b) Fails closed	Mechanical binding	Unable to take H_2 and O_2 readings and to isolate gas grab sample.	Failure of both H_2 and O_2 analyzers to indicate when recirculating sample.	None	See 5) - a)

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Table 9.3.16

POST ACCIDENT SAMPLING SYSTEM (PASS)
FAILURE MODE AND EFFECTS ANALYSIS (Sheet 23 of 38)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
70	Containment atmosphere line temperature indicator; T-503	a) false indication of high temperature	Electro-mechanical failure	Possible condensation in containment mechanical atmosphere lines of steam from containment.	Possible fluctuations in flow, F-503 if line becomes plugged with water.	None	See 70) - a)
		b) false indication of low temperature	Electro-mechanical malfunction	Possible overheating of line due to heat tracing.	Operator	None	Heat tracing is only required following a steam release to the containment.
71	Normal Sample diverter valves; 2HV0518B 2HV0584 2HV0519B 2HV0582 2HV0588A	a) fails open	Mechanical binding. Loss of air or power.	No impact on normal plant operations. Flow to PASS goes through normal sample heat exchanger.	Operator (position indication at local control panel). Low flow, F-502, when purging the PASS	Downstream series isolation valves can be closed in the existing sampling system. Cooling water to heat exchanger can be valved out.	-
		b) fails closed	Mechanical binding	No impact on PASS normal system operations.	None in PASS. Low flow in existing sampling system if attempting to sample.	None req'd	-
72	Liquid Radwaste Header test connections; 1"-605-J-376 1"-606-C-376	a) fails open	Contamination, mechanical damage	Possible loss of sample fluid either inside containment through 1"-605-J-376 or outside containment through 1"-606-C-376.	Operator	Drain line is capped to prevent leakage.	Valve is normally closed.
		b) fails closed	Mechanical binding	No impact on normal system operations. Unable to test containment isolation valves in sample return line to RDT.	Operator	None	-

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Table 9.3.16

POST ACCIDENT SAMPLING SYSTEM (PASS)
FAILURE MODE AND EFFECTS ANALYSIS (Sheet 24 of 38)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
73	Containment atmosphere inlet isolation valve; 2HV-7816	a) fails open	Mechanical binding	No impact on normal PASS system operations.	None	The PASS design pressure can accommodate maximum containment pressures.	-
		b) fails closed	Mechanical binding. Loss of air or power.	Unable to obtain a containment atmosphere sample in the PASS.	Power indication at the control panel if failure due to loss of power. Low flow, F-503, during sampling.	None	Reactor coolant sample analyses is not affected by this failure.
74	Containment atmosphere outlet isolation valve; 2HV-7261	a) fails open	Mechanical binding	No impact on normal PASS system operations.	None	PS-238 provides redundant isolation for the PASS containment sample line.	-
		b) fails closed	Mechanical binding. Loss of air or power.	Unable to obtain containment atmosphere sample with PASS. Unable to purge reactor coolant gas lines to containment after sampling.	Power indication at the control panel if failure due to loss of power. Low flow F-503, during sampling.	None	Boron, pH, local liquid grab sample and liquid isotope readouts not affected by this failure.
75	PASS sample inlet isolation valves; 2HV0519A 2HV0518A	a) fails open	Mechanical binding	No impact on PASS normal operations.	None	Redundant upstream isolation is provided from each source of sample water.	-
		b) fails closed	Mechanical binding. Loss of air or power.	Unable to obtain a liquid sample from the sample line with the failed valve.	Power indications at the control panel if failure due to loss of power. Low flow, F-502, if sampling from the line when the failure occurs.	None	Containment atmosphere sampling and sampling from the functioning inlet line are not affected by this failure.

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Table 9.3.16

POST ACCIDENT SAMPLING SYSTEM (PASS)
FAILURE MODE AND EFFECTS ANALYSIS (Sheet 27 of 38)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
82	RCDT Header isolation valve; 2HV7532B	a) fails open	Mechanical binding	No impact on normal system operation. Possibly unable to discharge from PASS if RCDT is at a higher pressure than the PASS.	Low flow, F-502, in conjunction with high RCDT pressure.	None	See 80) - b)
		b) fails closed	Mechanical binding. Loss of power.	No impact on normal system operations since PASS discharge can be diverted to containment sump.	Power indication at the control panel if failure due to loss of power. No flow, F-502, if sampling in progress.	2HV7532A provided an alternate drain line to the containment sump.	-
83	PASS containment sump isolation valve; 2HV7532A	a) fails open	Mechanical binding	No impact on normal system operations. PASS discharge flow diverted to containment sump. Possible blowdown of RCDT to sump of PS-231 open.	Possible decreasing level RCDT.	2HV7532C provides isolation capability from PASS if desired.	-
		b) fails closed	Mechanical binding. Loss of power.	Unable to discharge to containment sump. Possible alternate drain path RCDT, if RCDT pressure is lower than PASS pressure.	Power indication at the control panel if failure due to loss of power. No flow, F-502, if discharging to containment sump.	None	Discharge flow is normally diverted to the RCDT.

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Table 9.3.16

POST ACCIDENT SAMPLING SYSTEM (PASS)
FAILURE MODE AND EFFECTS ANALYSIS (Sheet 28 of 38)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection	Inherent Compensating Provision	Remarks and Other Effects
84	PASS strainer	a) plugged	Containment buildup	Reduced PASS sample flow.	Low flow, F-502.	Strainer can be bypassed while strainer maintenance is performed. Remote blow-down of strainer through HV0536 should clear strainer.	No affect on normal sampling capabilities once strainer is blown down.
		b) fails to strain properly	Element "punch through", wrong size element	Possible deposition of particulates in the system components.	Possible low flow, F-502.	Strainer can be bypassed while strainer maintenance is performed.	Strainer is only required when sampling from the containment sump.

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