DEC 9 1981

Docket No.: STN 50-447

General Electric Company
ATTN: Glenn G. Sherwood, Manager
Safety and Licensing
Operation
Nuclear Reactor Systems Division
175 Curtner Avenue, Mail Code 682
San Jose, California 95125

Dear Mr. Sherwood:

Subject: Acceptance Review of Application for Final Design Approval for 238 Nuclear Island

We have completed our acceptance review of the Standard Safety Analysis Report, GESSAR II, of your tendered application for final design approval for the 238 Nuclear Island. As a result, we have concluded that GESSAR II, subject to the comments provided below, is sufficiently complete to permit us to initiate our detailed review. It should be noted that substantive deficiencies may exist in some sections that will need to be corrected.

Accordingly, your filing of the application should include three (3) originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, your filing should include fifteen (15) copies of the general information attachment and forty (40) copies of the Standard Safety Analysis Report. As required by Section 50.30 of 10 CFR Part 50, you should retain an additional ten (10) copies of the general information attachment and thirty (30) copies of the Standard Safety Analysis Report for direct distribution in accordance with instructions which might be provided later. For all subsequent amendments to the Standard Safety Analysis Report sixty (60) copies will be required for distribution.

In addition to the generic information applicable to all potential applicants referencing GESSAR II, some unique facility information is provided in the tendered SSAR. This information is enclosed in boxes, and the pages on which such information is presented are a different color than the remaining pages of the SSAR. We believe that this arrangement may be confusing to technical reviewers. Consequently, please submit your docketed version

OFFICE ▶							
SURNAME ▶		8112290515 811209 PDR ADOCK 05000447 A PDR					
DATE ▶							

without facility unique information, indicating that such information will be supplied by the applicant. In making this adjustment, there is no longer a need for using colored pages.

During the course of our acceptance review of the SSAR, we identified a number of areas where additional information will be required for us to perform a detailed review. These are discussed in Enclosure 1 as a request for additional information. We request that you amend your SSAR to include the requested information in Enclosure 1 within three months from the docketing date.

We note that the attachment to your letter of March 31, 1980, that submitted GESSAR II for acceptance review, indicates that TMI matters will be addressed in NEDO-25224. Subsequent to the issuance of NEDO-25224, the Commission approved a list of TMI-related requirements, provided in NUREG-0737, "Clarification of TMI Action Plan Requirements." Therefore, for us to perform a review of the 238 Nuclear Island with regard to TMI requirements, GESSAR II should be amended to address conformance of the plant to all of the applicable requirements contained in NUREG-0737. Also, NUREG-0737 provides direction on the timing for submittal of information and documentation relating to the implementation of the TMI-related requirements. We request that within three months from the docketing date, you submit an amendment to the SSAR which includes the TMI-related information that can be provided at the time, and a schedule, consistent with the direction given in NUREG-0737, as to when the remaining TMI-related information will be provided.

In addition to the TMI-related requirements, there are other review areas in which requirements have been added or modified, or in which staff concerns have been raised in the review of other pending applications. A number of these areas are identified in Enclosure 2, and guidance on these areas is provided in Enclosures 3 through 12. To expedite the review process for your application, we request that you evaluate these areas and, where appropriate, upgrade your SSAR to include how these requirements are met or how these staff concerns are resolved for your nuclear island design. We request that within thirty days of docketing you provide us with a schedule for providing the remaining applicable information. In providing your schedule, we are assuming that the General Electric Company is willing to commit the necessary resources to complete the GESSAR II review in a timely manner.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Following docketing, we plan to develop a review schedule for the 238 Nuclear Island application which will be based on the assumption that the additional information requested in Enclosures 1 and 3 through 12 are provided within the times specified. If you cannot meet these dates or if you have any questions about the planned procedural aspects for the GESSAR II review, please advise us so that we can appropriately reflect those considerations into the development of the review schedule. After the schedule has been developed, you will be advised of the key milestones of the review.

If during the course of our review you believe that there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that an appropriate meeting can be arranged on a timely basis. A written request is not necessary; all such requests should be initiated through Howard Faulkner the licensing project manager assigned to the review of your application. This procedure is an informal one, designed to allow opportunity for the applicant to discuss with NRC management, areas of disagreement in the case review.

The reporting requirements contained in this letter affect fewer than ten respondents and, therefore, are not subject to OMB clearance as required by P.L. 96-511.

Sincerely,

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As Stated

cc w/enclosures:
See next page

DISTRIBUTION:

Docket Files	H. Faulkner
NRC PDR	J. Knight
Local PDR	W. Johnston
TERA	D. Muller
NSIC	P. Check
TIC	W. Kreger
SSPB Reading	L. Rubenstein
D. Eisenhut	J. Kramer
R. Purple	F. Schroeder
R. Tedesco	M. Ernst
J. Miller	OELD
LA	

*SEE PREVIOUS CONCURRENCE

OFFICE	DL:SSPB	DL:SSPB*	OELD*	DL:AD/L	DL:DD	DL:DIR	
SURNAME	HFaulkner/cc	JMiller	BBordenick	RTedesco	RPurple	DEisenhut	
DATE	10/22/81	10/ /81	10/ /81	10/28/81	10/22/81	10/ /81	

consistent with any direction given for submittal in the discussion of these review areas, as to when the remaining applicable information will be provided.

Following docketing, we plan to develop a review schedule for the 238 Nuclear Island application which will be based on the assumption that the additional information requested in Enclosures 1 and 3 through 12 are provided within the times specified. If you cannot meet these dates or if you have any questions about the planned procedural aspects for the GESSAR II review, please advise us so that we can appropriately reflect those considerations into the development of the review schedule. After the schedule has been developed, you will be advised of the key milestones of the review.

If during the course of our review you believe that there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that an appropriate meeting can be arranged on a timely basis. A written request is not necessary; all such requests should be initiated through Howard Faulkner the licensing project manager assigned to the review of your application. This procedure is an informal one, designed to allow opportunity for the applicant to discuss with NRC management, areas of disagreement in the case review.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

cc w/enclosures: See next page

Enclousres:
As Stated

DISTRIBUTION:

Docket Files (DCS-016)	H. Faulkner
NRC PDR	JKnight
Local PDR	WJohnston
TERA	DMuller
NSIC	PCheck
TIC	WKreger
SSPB Reading	LRubenstein
D. Eisenhut	JKramer
R. Purple	FSchroeder
R. Tedesco	MErnst
J. Miller	OELD
L. Tremper	

OFFICE	DL:SSPB	DL:SSPB	DL:AD/L	DL:DD	DL:DIR	OELD
SURNAME	HFaulkner/cc	JMiller	RTedesco	RPurple	DEisenhut	BBordenick
DATE	8/18/81	8/20/81	8/31/81	8/ /81	8/ /81	8/ /81

General Electric Company
ATTN: Glenn G. Sherwood, Manager
Safety & Licensing Operation
Nuclear Power Systems Division
175 Curtner Avenue, Mail Code 682
San Jose, California 95125

cc: Mr. Joseph F. Quirk, Manager
BWR Standardization
General Electric Company
175 Curtner Avenue
San Jose, CA 95114

Mr. L. Gifford, Manager
Regulatory Operations Unit
General Electric Company
4720 Montgomery Lane
Bethesda, Maryland 20014

Director, Criteria & Standards Division
Office of Radiation Programs
U.S. Environmental Protection Agency
401 M Street, S.W.
Washington, D.C. 20460

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION
STANDARD SAFETY ANALYSIS REPORT
GESSAR II

Docket No. STN 50-447

100 MISCELLANEOUS

100.1 (1.3) Per Regulatory Guide 1.70, reference the sections of the SAR that describe the GESSAR parameters.

100.2 (13.0) Chapter 13 is missing. If the applicant will supply Chapter 13, please state.

100.3 (SSAR) Many figures are difficult to read or are illegible. The quality of these figures should be improved.

210.0 MECHANICAL ENGINEERING

210.1 (3.6.2.2) Supply the missing information in this section.

210.2 (3.9.3) Supply the missing information in Tables 3.9-10 and 3.9-11.

220.0 STRUCTURAL ENGINEERING

220.1 (3.7.2.6)	Indicate the extent to which the recommendations of Regulatory Guide 1.92 are followed.
220.2 (3.7.2.7)	Indicate the extent to which the recommendations of Regulatory Guide 1.92 are followed.
220.3 (3.7.4)	Supply the missing information in Table 3.7-53.
220.4 (3.8.2.4)	Supply the missing information for Table 3.8-3 and Figure 3.8-7.
220.5 (3.8.2.5)	Indicate the extent to which the recommendations of Regulatory Guide 1.57 are followed.
220.6 (3.8.2.6)	The quality control program that is proposed for the fabrication and construction of the containment should be described with emphasis on the extent of compliance with Article NE-5000 of the ASME Code, Section III, Division 1, including welding procedures and erection tolerances.
220.7 (3.8.3.3, 3.8.3.4, & 3.8.3.6)	Indicate the extent to which compliance with the criterion of ACI-349 is accomplished.
220.8 (3.8.3.4)	Indicate the extent to which the recommendations of Regulatory Guide 1.57 are followed.

241.0 GEOTECHNICAL ENGINEERING

241.1 (2.5.1) Provide details of the broad spectrum of foundation conditions that have been used to arrive at the design loads. Discuss the procedure used to select the foundation conditions and the method of computing the design loads. Present the information in the appropriate sections of the SSAR.

241.2 (3.7.1.1) You mention that the bases for selection of design spectra are discussed in Section 3.7.2.5. The section number appears to be incorrect. Provide the correct reference.

241.3 (3.7.1.4.1) Explain how the finite-element representation has been used to model all the supporting medium conditions. Provide appropriate figures, if necessary.

241.4 (3.7.1.4.2) Explain and justify the use of a single curve representative of sandy soil properties for representing other sands, clays and silty soils that may be encountered at various sites. Provide this information in Section 3.7.1.3 of the SSAR.

241.5 (3.8) Based on a review of the range of soil properties used in the GESSAR II seismic analysis, we do not find an adequate basis to agree that the "uncertainties in soil properties and frequencies are adequately accounted for in the envelope design." We believe that the envelope design will meet all the requirements of a specific design based on site specific geotechnical parameters, and the parameters will be based on state-of-the-art soil exploration and seismic analysis for a specific site.

241.6 (3.8.2.3.9) Describe in detail the procedure followed to arrive at the soil properties used in the model for LOCA/SRV loads analysis.

241.7 (3.8.4) Provide a plan and profile of Category I pipelines that will be buried in soil.

241.8 (3.8.5.4) Discuss how the effects of soil and structural settlement are considered, and state the acceptance criteria for the proposed values.

241.9 (3.8.5.4.1) Describe in detail the procedure used for calculating the subgrade stiffness used in NASTRAN. Discuss the applicability of the stiffness values to various sites with different soil conditions.

241.10 (3.8.5.4.1) Your reference to Subsection 3.7.14 is incorrect. Provide the correct reference.

241.11 (3.8.5.4.1) The factors of safety against sliding given in Figure 3.8-78 for the reactor building, auxiliary building and control building are below those of the Standard Review Plan (Section 3.8.5). Justify the use of such low factors of safety.

241.12 (3.8.6.2) Describe how the ultimate and residual soil settlements were calculated, and discuss the applicability of these computations to a range of site conditions. Provide the required orientation, location and purpose of settlement points on Figure 3.8-91. Explain how settlement values will be interpreted, and establish limiting criteria.

241.13 (3.8.6.2) Clarify the meaning of the last sentence of subsection 3.8.6.2, which states that the actual soil properties will be compared with the required soil properties in the building design stress reports. Describe how the required soil properties will be determined and how comparisons will be made.

241.14 (3A.1.2) (PSP) The staff does not agree that site-unique seismic analysis or review by regulatory agencies should not be required. It is the position of the staff that the applicant must demonstrate that, based on actual geotechnical site parameters and the state-of-the-art at the time of submission of an FSAR, the seismic analysis results given in GESSAR II envelope the results of the seismic analysis for actual sites.

241.15 (3A.1.2) State that the applicant will investigate the liquefaction potential of the foundation and site soils for a long duration, New Madrid type earthquake.

241.16 (3A.1.2) Omit the name of the Branch of the NRC staff that will review the analysis.

241.17 (3A.1.2) You state that in your analysis no special restrictions were provided for parameters, such as variation of water table, material density, material composition or soil profile. Discuss how your analysis is applicable to many potential sites in light of this lack of parameter variation. In the seismic analysis for layered soil sites, not only the range of parameters for soil properties is important, but also the sequence in which the soil layering exists. Explain how soil layering is accounted for in your analysis. The water table elevation not only affects strain dependent material properties for sandy soils, but also the compressional wave velocity for the soil-structural interaction analysis due to vertical excitation. Justify in detail your approach to these items.

241.18 (3A.2.2) Your description of the soil damping curve used in the analysis is different than that presented in Subsection 3.7.1.4.2. The two descriptions should be consistent. Describe how the damping properties for clays, tills, or other materials have been accounted for. Also, indicate how you plan to justify your damping curve on a site-specific application.

241.19 (3A.2.2) Most of the soil profiles, other than fixed base conditions, that you have analyzed are 75 ft. deep. What is the bases for assuming this profile represents a wide range of soil profile conditions?

241.20 (3A.2.2) Your selection of shear wave velocity profiles does not seem to include a wide range of soil profiles. Your lower bound soil properties are very stiff below the foundation elevation of the reactor building. Other profiles are much stiffer and many of these are close to being representative of rock sites. Please justify your selections.

241.21 (3A.2.2) Figure 3A-5 shows shear wave velocity profiles to a depth of 300 ft. Most of your profiles, other than fixed base, are 75 ft. deep. One profile is 150 ft. deep. What values of shear wave velocities were used below the depth of the analyzed profile, and what was the basis for selecting these values?

241.22 (3A.3.1) Provide velocity and displacement time histories corresponding to input acceleration time histories to illustrate that the time histories used in the analysis are base line corrected.

241.23 (3A.3) Provide details of the interpolation control number scheme used for the solution in FLUSH.

241.24 (3A.5.2) For vertical SSI analysis, the depth of the water table governs the compressional wave velocity in the medium. Reevaluate your results for vertical analysis taking into account the effect of the depth of the water table.

241.25 (3A.5.2) You have made a vertical analysis for two profiles. In Figure 3A-24, you have shown nine response spectra. What do these spectra plots represent? Provide a description and table that relates the response spectra plots to the profiles.

241.26 (3A.5.2) You have presented the results of free field and interaction response spectra at basemat level for the OBE only. Provide these results for the SSE also.

251.0 MATERIALS ENGINEERING - COMPONENT INTEGRITY

251.1 (3.5.1.3) Provide or reference the analyses for both high and low trajectory missiles that demonstrate the conclusions in this section.

270.0	<u>ENVIRONMENTAL QUALIFICATION</u>
270.1 (3.11.1 & 3.11.2)	Supply the missing information in Tables 3.11-2, 3.11-3, 3.11-8 and 3.11-9.
270.2 (3.11.2)	Indicate how the requirements of GDC-50 of Appendix A to 10 CFR Part 50 have been met.
270.3 (3.11.2)	Indicate the extent of compliance with NUREG-0588.
270.4 (3.11.4)	Supply the missing information in this section.

271.0 SEISMIC AND DYNAMIC LOAD QUALIFICATION

271.1 (3.10) Supply the missing information in Table 3.10-1.

280.0 FIRE PROTECTION

280.1 (9.5.1.3) As per Regulatory Guide 1.70, a failure mode and effects analysis should be provided that demonstrates that operation of the fire protection system in areas containing engineered safety features would not produce an unsafe condition or preclude safe shutdown. The effects of firefighting activities and fire suppression agents on safety systems should be discussed. An analysis of the fire detection and protection system with regard to design features to withstand the effects of single failures should be included.

410.0 AUXILIARY SYSTEMS

410.1 (9.1.2.3) As per Regulatory Guide 1.70, include a discussion of material compatibility requirements in the safety evaluation.

410.2 (9.1.4.2) Either provide a description of the containment polar crane or indicate this will be provided by the applicant. Provide any fuel handling system interface information as per Regulatory Guide 1.70, Revision 3.

410.3 (9.1.4.3) As per Regulatory Guide 1.70, the results of a failure mode and effects analysis should be presented to demonstrate that the individual subsystems and components, including controls and interlocks, are designed to meet the single-failure criterion without compromising the capability of the system to perform its safety function.

410.4 (9.1.4.3) Indicate the portion of this section to be supplied by the applicant (i.e., demonstrate that failure of any part of the spent fuel cask-handling crane will not cause any damage to spent fuel and safety-related equipment) or provide the missing information.

410.5 (9.1.4.5) The adequacy of safety-related interlocks to meet the single-failure criterion should be demonstrated.

410.6 (9.2.1) Designate the responsibility and requirements, as appropriate, for demonstrating the capability of the system to function during abnormally high and low water levels and for preventing organic fouling that may degrade system performance.

410.7 (9.2.6) Provide a discussion of the environmental design considerations, requirements for leakage control (including mitigation of environmental effects) and limits for radioactivity concentration. Also, provide an analysis of storage facility failure provisions for mitigating environmental effects, and an evaluation of radiological considerations (to be presented in Chapter 12). Indicate the areas of applicant responsibility.

410.8 (9.3.5.3) The results of a failure mode and effects analysis should be presented to demonstrate that the system can meet the single-failure criterion without compromising the shutdown capability of the system. (The reference to Section 15A.6.6 is not adequate.)

410.9 (9.4.2.1) As per Regulatory Guide 1.70, include requirements for meeting the single-failure criterion, seismic design criteria, requirements for the manual or automatic actuation of system components or isolation dampers, preferred direction of airflow from areas of low potential radioactivity to areas of high potential radioactivity, monitoring normal and abnormal radiation levels within the area, differential pressures to be maintained and measured, and the requirements for treatment of exhaust air. Details of the means for protection of system vents or louvers from missiles should be provided.

410.10 (9.4.3.1) As per Regulatory Guide 1.70, include any requirements for preferred direction of airflow from areas of low potential radioactivity to areas of high potential radioactivity, differential pressures to be maintained and measured, requirements for the monitoring of normal and abnormal radiation levels, and requirements for the treatment of exhaust air.

410.11 (10.3) For each of the sub-paragraphs listed in Regulatory Guide 1.70, Section 10.3, Revision 3, list the corresponding section in GESSAR II, Section 5.4, that provides the required information. Also, identify those sub-paragraphs in Regulatory Guide 1.70, Section 10.3, Revision 3, where information will be provided by the applicant.

420.0	<u>INSTRUMENTATION AND CONTROL SYSTEMS</u>
420.1 (7.0)	Throughout Chapter 7.0 you refer to Figures 7A.X.X (See Table 7.8-1 in Section 7.8, Page 7.8-3 for an example). Please clarify these references, since these figures cannot be found in the SSAR.
420.2 (7.1.1.7)	Provide the information identified as "later."
420.3 (7.2.2)	As per Regulatory Guide 1.70, include a failure mode and effects analysis.
420.4 (7.2.2.1)	The text identifies more references than are given in Section 7.2.3, References. Provide appropriate bibliography for all references.
420.5 (7.2.2.1)	Pages 7.2-58 and -59 contain erroneous references to Section 7.2.2.1 for additional discussion. Correct or clarify these references.
420.6 (7.3)	Provide the text, tables and figures identified as "later."
420.7 (7.3.2)	As per Regulatory Guide 1.70, include the failure mode and effects analyses.
420.8 (7.4.1)	Specifically address the design basis information required by Section 3 of IEEE 279-1971.
420.9 (7.4.1.1)	Provide the missing information in Figures 7.4-1a/1b/1c/1d and 7.4-2.
420.10 (7.4.1.2)	State specifically whether you comply fully with NRC General Design Criterion 19.

420.11 (7.5.1.1) Provide the missing information indicated in Table 7.5-1 and Figures 7.5-1 and 7.5-5. In addition, provide instrumentation accuracy in Table 7.5-1.

420.12 (7.6) Provide the missing information in Figures 7.6-1, 7.6-12 and 7.6-17 and in Tables 7.6-2 and 7.6-3.

430.0 POWER SYSTEMS

430.1 (8.3.1.1.5.1) Provide Figure 8.3-5.

430.2 (9.5.2.2.1) Provide Figures 9.5-6 and 9.5-9.

430.3 (9.5.8.1) Either provide or indicate the applicant will provide the design basis regarding contaminating substances as related to the facility site, systems and equipment.

440.0 REACTOR SYSTEMS

440.1 (5.2.2.2.2.1) Section 5.2.2.2.2.1 provides a listing of the most severe operating conditions. Item (2) of this section does not list an assumed pressure. Provide the missing information.

440.2 (5.2.2.7) Section 5.2.2.7 references Figure 5.2-5. Please correct the SSAR, since, apparently, this reference should be Table 5.2-5.

440.3 (15.6.5) Address each item identified in Item 1 of Table 15-4 of Regulatory Guide 1.70, Revision 3, or indicate an interface to provide the information.

440.4 (15.0.3.3.2, 15.2.9.3.3, 15.6.4.3.1, 15.6.5.3.2, 15.7.1.1.5.2.1.2) For the reactor system parameters identified in the referenced sections of Chapter 15, specify the permitted operating band (permitted fluctuations in a given parameter and associated uncertainties). Confirm that the most adverse conditions within the operating band are used as initial conditions for the transient analysis.

450.0 ACCIDENT EVALUATION

450.1 (15.7.1.1.5) In the evaluation of the radiological results, consider uncertainties in calculational methods, equipment performance, instrumentation response characteristics or other indeterminate effects.

450.2 (15.7.1.1.5) Address each item identified in Item 2 of Table 15-4 of Regulatory Guide 1.70, Revision 3, or indicate an interface to provide the information specified therein.

450.3 (15.7.2) Provide a sequence of events that includes the means of operator recognition and diagnosis of equipment/component failure. Describe operator actions.

450.4 (15.7.2.5) Describe or reference the computational methods for determining the values in Tables 15.7-8 and 15.7-10.

450.5 (15.7.4.3) Provide the following information:

a. Pool decontamination factors
b. Maximum fuel rod pressurization
c. Minimum water depth between top of fuel rods and fuel pool surface
d. Peak linear power density for the highest power assembly discharged
e. Maximum centerline operating fuel temperature for the fuel assembly in item (d) above
f. Average burnup for the peak assembly in item (d) above.

460.0 EFFLUENT TREATMENT SYSTEMS

460.1 (11.2.1) Address and describe how the requirements of GDC 60 and 64 of Appendix A to 10 CFR 50 are achieved and will be implemented relative to liquid wastes.

460.2 (11.2.3) Provide the bases for the values in Table 11.2-4.

460.3 (11.3.2.16) Tabulate or indicate whether the applicant will supply calculated concentrations of airborne radioactive material (by radionuclide) expected during normal and anticipated operational occurrences for equipment cubicles, corridors and areas normally occupied by operating personnel.

460.4 (11.3.4) Tabulate or indicate whether the applicant will provide the releases by radionuclide for each subsystem and for the total system, and indicate the effluent concentrations. The calculated effluents should be compared with the concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 1; the doses due to the effluents should be compared with the numerical design objectives of Appendix I to 10 CFR Part 50 and the dose limits of 10 CFR Part 20.

460.5 (11.3.4.1) Identify or indicate whether the applicant will provide all release points of gaseous waste to the environment on process flow diagrams, general arrangement drawings, or a site plot plan.

For release points, give:

1. Height of release,
2. Inside dimensions of release point exit,
3. Effluent temperature, and
4. Effluent exit velocity.

460.6 (11.4.1.2) Indicate and address how the requirements of 10 CFR 50 will be met.

460.7 (11.4.2) Provide the following information: 1) piping and instrumentation diagrams that show system interconnections and quality group interfaces; 2) a description of the design provisions incorporated to control the release of radioactive materials due to overflow from tanks containing liquids, sludges, and spent resins; 3) a discussion of the methods of handling and packaging large waste materials and equipment that has been activated during reactor operation (e.g., core components).

460.8 (11.4.2.3.2) Provide a tabulation (for the dry solid-waste subsystem) of the maximum and expected waste inputs in terms of type (dry ventilation filters, contaminated clothing, equipment, tools, etc.), sources of waste, volume and isotopic and curie content and, also give the bases for the values.

471.0 RADIATION PROTECTION

471.1 (12.2.1) Indicate whether, and if so, how, the applicable guidance provided in ANSI N237 has been followed; if not followed, describe the specific alternative method used.

471.2 (12.2.1.2) Provide the missing information in Tables 12.2-6, 12.2-7, 12.2.-16 and 12.2-18.

471.3 (12.3.1) In Section 12.3.1 relative to Figures 12.3-8 through 12.3-24, show or reference shield wall thickness, and the location of airborne radioactivity and area radiation monitors.

471.4 (12.3.3) Address or indicate whether the applicant will provide the criteria established for the changeout of air filters and absorbers in the air cleaning system. Indicate whether, and if so, how, the applicable guidance provided in Regulatory Guide 1.52 has been followed; if not followed, describe the specific alternative method used.

480.0 CONTAINMENT SYSTEMS

480.1 (6.2.2.3) Per the requirements of Regulatory Guide 1.70, Section 6.2.2.3, provide a failure mode and effects analysis of the containment heat removal systems.

492.0 CORE PERFORMANCE - THERMAL HYDRAULICS

492.1 (4.4.2.1) Per Regulatory Guide 1.70, the thermal and hydraulic design parameters of the reactor are to be provided in a summary and compared to similarly designed reactors. Table 4.4-1 provides a summary, but comparison data [illegible] not supplied.

640.0 PROCEDURES AND TEST REVIEW

640.1 (14.2.5) State clearly whose approval must be obtained before increasing power to the next higher test plateau.

ENCLOSURE 2

ADDITIONAL GUIDANCE FOR GESSAR II

DOCKET NO.: STN 50-447

(1) Environmental Qualification of Safety Related Electrical Equipment - Commission Memorandum and Order of May 23, 1980 defines the current staff requirements for qualification of this equipment. Additional guidance on this matter was provided in a subsequent NRR Order, dated November 26, 1980 (concerning record requirements), Supplements 2 and 3, dated September 30, 1980 and October 24, 1980, respectively, to IE Bulletin No. 79-01B, and a generic letter to all holders of CPs and OLs, dated October 1, 1980.

(2) Seismic Qualification - A staff request for additional information in this review area has been sent to a number of pending OL applicants. A copy of that request is provided as Enclosure 3.

(3) Fire Protection - The current requirements for the fire protection programs are defined in the new Appendix R to 10 CFR Part 50. As further guidance, a copy of the staff position and a recent staff request for additional information are provided in Enclosure 4. (Not all questions need to be answered as some are redundant. However, your submittal should address all these items.)

(4) Masonry Walls - The staff requirements regarding this issue are stated in Appendix A to Standard Review Plan 3.8.4, Interim Criteria for Safety-Related Masonry Wall Evaluation. A copy of Appendix A is provided as Enclosure 5.

(5) Fracture Prevention of Containment Pressure Boundary (GDC 51) - Enclosure 6 provides clarification on how the staff determines compliance with GDC 51.

(6) Initial Test Program Descriptions (Chapter 14) - Staff review of near term OL applications has revealed a number of concerns which are common to pending applications. The nature of these concerns are typically expressed in the questions the staff has raised in its review of the Summer and the San Onofre 2 & 3 applications.

(7) Special Low Power Test Program (Task Action Plan Item I.G.1) - The staff has established guidance for this matter for transmittal to all pending and prospective BWR OL applicants. A copy of that guidance is provided as Enclosure 7.

(8) Preservice and Inservice Inspections - Staff guidance in this review area has been sent to a number of pending OL applicants. A copy of that guidance is provided as Enclosure 8.

(9) Preservice Inspection & Testing of Snubbers - The staff has recently established requirements to ensure snubber operability which have been transmitted to pending OL applicants. A copy of those requirements is provided as Enclosure 9.

(10) Effects of Containment Coatings and Sump Debris on ECCS and Containment Spray Operation - A copy of the NRC staff concerns on this issue, including a request for additional information which has been sent to a number of OL applicants, is provided as Enclosure 10.

(11) Instrumentation for Detection of Inadequate Core Cooling (TMI Action Item II.F.2 in NUREG-0737) - Discussion of this item should address how core thermocouple readouts are provided in the control room including location and rate of printout (see Part (4) of attachment 1 to Item II.F.2).

(12) Safety - Related Structures, Systems and Components (Q-List) Controlled by the QA Program - Staff requests for additional information regarding this issue have been sent to a number of OL applicants. A recent request regarding Diablo Canyon is provided as Enclosure 11.

(13) Tornado Missile Protection with Regard to Ventilation Openings in Buildings Housing Essential Shutdown Equipment - A discussion of the staff concern regarding this issue is provided in Supplement No. 5 to the SER for Sequoyah 1 & 2.

(14) Instrumentation and Control Systems - The staff has recently identified four concerns in this review area. A discussion of the concerns including a request for information to assist in their resolution is provided as Enclosure 12.

Equipment Qualification Branch
Seismic Qualification Review Team
Request for Additional Information

ENCLOSURE 3

1. In accordance with the requirements of GDC 2 and 4 all safety-related equipment is required to be designed to withstand the effects of earthquakes and dynamic loads from normal operation, maintenance, testing and postulated accident conditions. GDC 2 further requires that such equipment be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of earthquake loads.

   The criteria to be used by the staff to determine the acceptability of your equipment qualification program for seismic and dynamic loads are IEEE Std. 344-1975 as supplemented by Regulatory Guides 1.100 and 1.92, and Standard Review Plan Sections 3.9.2 and 3.10. State the extent to which the equipment in your plant meets these requirements and the above requirements to combine seismic and dynamic loads. For equipment that does not meet these requirements provide justification for the use of other criteria.

2. Provide a list of all safety-related systems together with a list of all safety-related equipment and support structures associated with each system. The equipment lists should indicate whether the equipment is NSSS supplied or BOP supplied. These lists should include all safety-related mechanical components, electrical, instrumentation, and control equipment, including valve actuators and other appurtenances of active pumps and valves.

3. For each safety-related equipment item, the following information should be provided:

   (1) Method of qualification used:

      a) Analysis or test (indicate the company that prepared the report, the reference report number and date of the publication).

      b) If by test, describe whether it was a single or multi-frequency test and whether input was single axis or multi-axis.

      c) If by analysis, describe whether static or dynamic, single or multiple-axis analysis was used.

      d) Provide natural frequency (or frequencies) of equipment.

   (2) Indicate whether the equipment has met the qualification requirements.

   (3) Indicate whether the equipment is required for:

      a) hot stand-by

      b) cold shutdown

      c) both

      d) neither

(4) Location of equipment, i.e., building, elevation.

(5) Availability for inspection (Is the equipment already installed at the plant site?)

(6) A compilation of the required response spectra (or time history) and corresponding damping for each seismic and dynamic load specified for the equipment together with all other loads considered in the qualification and the method of combining all loads.

4. Identify all equipment that may be effected by vibration fatigue cycle effects and describe the methods and criteria used to qualify this equipment for such loading conditions

5. Describe the results of any in plant tests, such as in situ impedance tests, and any plans for operational tests which will be used to confirm the qualification of any item of equipment.

6. To confirm the extent to which the safety-related equipment meets the requirements of General Design Criterion 2 and 4, the Seismic Qualification Review Team (SQRT) will conduct a plant site review. For selected equipment, SQRT will review the combined required response spectra (RRS) or the combined dynamic response, examine the equipment configuration and mounting, and then determine whether the test or analysis which has been conducted demonstrates compliance with the RRS if the equipment was qualified by test, or the acceptable analytical criteria if qualified by analysis.

   The staff requires that a "Qualification Summary of Equipment" as shown on the attached pages be prepared for each selected piece of equipment and submitted to the staff two weeks prior to the plant site visit. The applicant should make available at the plant site for SQRT review all the pertinent documents and reports of the qualification for the selected equipment. After the visit, the applicant should be prepared to submit certain selected documents and reports for further staff review.

Qualification Summary of Equipment

I. Plant Name: ______________________ Type:

1. Utility: ______________________ PWR__________

2. NSSS:__________ 3. A/E:__________ BWR__________

II. Component Name ______________________________________

1. Scope: [ ] NSSS [ ] BOP

2. Model Number: ______________________ Quantity:________

3. Vendor: ______________________________________

4. If the component is a cabinet or panel, name and model No. of the devices included: ______________________________

____

5. Physical Description a. Appearance______________________

b. Dimensions______________________________

c. Weight______________________________

6. Location: Building:______________________

Elevation:______________________

7. Field Mounting Conditions [ ] Bolt (No.____, Size____)
[ ] Weld (Length____)
[ ] ______________

8. a. System in which located:______________________

b. Functional Description:______________________

c. Is the equipment required for [ ] Hot Standby [ ] Cold Shutdown
[ ] Both [ ] Neither

9. Pertinent Reference Design Specifications:______________________

____

12/80

III. Is Equipment Available for Inspection in the Plant: [ ] Yes [ ] No

IV. Equipment Qualification Method:

[ ] Test [ ] Analysis [ ] Combination of Test and Analysis

Qualification Report*: ______________________________

(No., Title and Date) ______________________________

Company that Prepared Report: ______________________________

Company that Reviewed Report: ______________________________

V. Vibration Input:

1. Loads considered: a. [ ] Seismic only

   b. [ ] Hydrodynamic only

   c. [ ] Combination of (a) and (b)

2. Method of Combining RRS: [ ] Absolute Sum [ ] SRSS [ ] ____________ (other, specify)

3. Required Response Spectra (attach the graphs): ______________________________

4. Damping Corresponding to RRS: OBE ____________ SSE ____________

5. Required Acceleration in Each Direction: [ ] ZPA [ ] Other ____________ (specify)

   OBE S/S = ____________ F/B = ____________ V = ____________
   SSE S/S = ____________ F/B = ____________ V = ____________

6. Were fatigue effects or other vibration loads considered?

   [ ] Yes [ ] No

   If yes, describe loads considered and how they were treated in overall qualification program: ______________________________

   ____

   ____

*NOTE: If more than one report complete items IV thru VII for each report.

VI. <u>If Qualification by Test, then Complete*:</u>

1. [ ] Single Frequency [ ] Multi-Frequency: [ ] random
   [ ] sine beat
   [ ] ______________
2. [ ] Single Axis [ ] Multi-Axis
3. No. of Qualification Tests: OBE __________ SSE ________ Other ________ (specify)
4. Frequency Range: ____________________________________
5. Natural Frequencies in Each Direction (Side/Side, Front/Back, Vertical):

   S/S = ____________________ F/B = ________________ V = ________________
6. Method of Determining Natural Frequencies

   [ ] Lab Test [ ] In-Situ Test [ ] Analysis
7. TRS enveloping RRS using Multi-Frequency Test [ ] Yes (Attach TRS & RRS gra
   [ ] No
8. Input g-level Test: OBE S/S = _________ F/B = __________ V = ________

   SSE S/S = _________ F/B = __________ V = ________
9. Laboratory Mounting:

   1. [ ] Bolt (No. _____, Size _____) [ ] Weld (Length______) [ ]______
10. Functional operability verified: [ ] Yes [ ] No [ ] Not Applicable
11. Test Results including modifications made: ____________________________

    ____
12. Other test performed (such as aging or fragility test, including results):

    ____

    ____

*Note: If qualification by a combination of test and analysis also complete Item VII.

VII. If Qualification by Analysis, then complete:

1. Method of Analysis:

   [ ] Static Analysis [ ] Equivalent Static Analysis

   [ ] Dynamic Analysis: [ ] Time-History [ ] Response Spectrum

2. Natural Frequencies in Each Direction (Side/Side, Front/Back, Vertical):

   S/S = ________________ F/B = ________________ V = ________________

3. Model Type: [ ] 3D [ ] 2D [ ] 1D

   [ ] Finite Element [ ] Beam [ ] Closed Form Soluti

4. [ ] Computer Codes: ________________

   Frequency Range and No. of modes considered: ________________

   [ ] Hand Calculations

5. Method of Combining Dynamic Responses: [ ] Absolute Sum [ ] SRSS
   [ ] Other: ________________
   (specify)

6. Damping: OBE ________ SSE ______ Basis for the damping used: ________________

7. Support Considerations in the model: ________________

8. Critical Structural Elements:

A.

Identification	Location	Governing Load or Response Combination	Seismic Stress	Total Stress	Stress Allowable

B.

Max. Critical Deflection	Location	Maximum Allowable Deflection to Assure Functional Operability

STAFF POSITION
SAFE SHUTDOWN CAPABILITY

Staff Concern

During the staff's evaluation of fire protection progr ms at operating plants, one or more specific plant areas may be identified in which the staff does not have adequate assurance that a postulated fire will not damage both redundant divisions of shutdown systems. This lack of assurance in safe shutdown capability has resulted from one or both of the following situations:

* Case A: The licensee has not adequately identified the systems and components required for safe shutdown and their location in specific fire areas.

* Case B: The licensee has not demonstrated that the fire protection for specific plant areas will prevent damage to both redundant divisions of safe shutdown components identified in these areas.

For Case A, the staff has required that an adequate safe shutdown analysis be performed. This evaluation includes the identification of the systems required for safe shutdown and the location of the system components in the plant. Where it is determined by this evaluation that safe shutdown components of both redundant divisions are located in the same fire area, the licensee is required to demonstrate that a postulated fire will not damage both divisions or provide alternate shutdown capability as in Case B.

For Case B, the staff may have required that an alternate shutdown capability be provided with is independent of the area of concern or the licensee may have proposed such a capability in lieu of certain additional fire protection modifications in the area. The specific modifications associated with the area of concern along with other systems and equipment already independent of the area form the alternate shutdown capability. For each plant, the modifications needed and the combinations of systems which provide the shutdown functions may be unique for each critical area; however, the shutdown functions provided should maintain plant parameters within the bounds of the limiting safety consequences deemed acceptable for the design basis event.

Staff Position

Safe shutdown capability should be demonstrated (Case A) or alternate shutdown capability provided (Case B) in accordance with the guidelines provided below:

1. Design Basis Event

   The design basis event for considering the need for alternate shutdown is a postulated fire in a specific fire area containing redundant safe shutdown cables/equipment in close proximity where it has been determined that fire protection means cannot assure that safe shutdown capability will be pre..rved. Two cases should be considered: (1) offsite power is av..able; and (2) offsite power is not available.

2. Limiting Safety Consequences and Required Shutdown Functions

2.1 No fission product boundary integrity shall be affected:

a. No fuel clad damage;
b. No rupture of any primary coolant boundary;
c. No rupture of the containment boundary.

2.2 The reactor coolant system process variables shall be within those predicted for a loss of normal ac power.

2.3 The alternate shutdown capability shall be able to achieve and maintain subcritical conditions in the reactor, maintain reactor coolant inventory, achieve and maintain hot standby* conditions (hot shutdown* for a BWR) for an extended period of time, achieve cold shutdown* conditions within 72 hours and maintain cold shutdown conditions thereafter.

* As defined in the Standard Technical Specifications.

3. Performance Goals

3.1 The reactivity control function shall be capable of achieving and maintaining cold shutdown reactivity conditions.

3.2 The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWR's and in the pressurizer for PWR's.

3.3 The reactor heat removal function shall be capable of achieving and maintaining decay heat removal.

3.4 The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

3.5 The supporting function shall be capable of providing the process cooling, lubrication, etc. necessary to permit the operation of the equipment used for safe shutdown by the systems identified in 3.1 - 3.4.

3.6 The equipment and systems used to achieve and maintain hot standby conditions (hot shutdown for a BWR) should be (1) free of fire damage; (2) capable of maintaining such conditions for an extended time period longer than 72 hours if the equipment required to achieve and maintain cold shutdown is not available due to fire damage; and (3) powered by an onsite emergency power system.

3.7 The equipment and systems used to achieve and maintain cold shutdown conditions should be either free of fire damage or the fire damage to such systems should be limited such that repairs can be made and cold shutdown conditions achieved within 72 hours. Equipment and systems used prior to 72 hours after the fire should be powered by an onsite emergency power system; those used after 72 hours may be powered by offsite power.

3.8 These systems need not be designed to (1) seismic category I criteria; (2) single failure criteria; or (3) cope with other plant accidents such as pipe breaks or stuck valves (Appendix A BTP 9.5-1), except those portions of these systems which interface with or impact existing safety systems.

4. PWR Equipment Generally Necessary For Hot Standby

(1) Reactivity Control

Reactor trip capability (scram). Boration capability e.g., charging pump, makeup pump or high pressure injection pump taking suction from concentrated borated water supplies, and letdown system if required.

(2) Reactor Coolant Makeup

Reactor coolant makeup capability, e.g., charging pumps or the high pressure injection pumps. Power operated relief valves may be required to reduce pressure to allow use of the high pressure injection pumps.

(3) Reactor Coolant System Pressure Control

Reactor pressure control capability, e.g., charging pumps or pressurizer heaters and use of the letdown systems if required.

(4) Decay Heat Removal

Decay heat removal capability, e.g., power operated relief valves (steam generator) or safety relief valves for heat removal with a water supply and emergency or auxiliary feedwater pumps for makeup to the steam generator. Service water or other pumps may be required to provide water for auxiliary feed pump suction if the condensate storage tank capacity is not adequate for 72 hours.

(5) Process Monitoring Instrumentation

Process monitoring capability e.g., pressurizer pressure and level, steam generator level.

(6) Support.

The equipment required to support operation of the above described shutdown equipment e.g., component cooling water service water, etc. and onsite power sources (AC, DC) with their associated electrical distribution system.

5. PWR Equipment Generally Necessary For Cold Shutdown*

(1) Reactor Coolant System Pressure Reduction to Residual Heat Removal System (RHR) Capability

Reactor coolant system pressure reduction by cooldown using steam generator power operated relief valves or atmospheric dump valves.

(2) Decay Heat Removal

Decay heat removal capability e.g., residual heat removal system, component cooling water system and service water system to removal heat and maintain cold shutdown.

(3) Support

Support capability e.g., onsite power sources (AC & DC) or offsite after 72 hours and the associated electrical distribution system to supply the above equipment.

* Equipment necessary in addition to that already provided to maintain hot standby.

6. BWR Equipment Generally Necessary For Hot Shutdown

(1) Reactivity Control

Reactor trip capability (scram).

(2) Reactor Coolant Makeup

Reactor coolant inventory makeup capability e.g., reactor core isolation cooling system (RCIC) or the high pressure coolant injection system (HPCI).

(3) Reactor Pressure Control and Decay Heat Removal

Depressurization system valves or safety relief valves for dump to the suppression pool. The residual heat removal system in steam condensing mode, and service water system may also be used for heat removal to the ultimate heat sink.

(4) Suppression Pool Cooling

Residual heat removal system (in suppression pool cooling mode) service water system to maintain hot shutdown.

(5) Process Monitoring

Process monitoring capability e.g., reactor vessel level and pressure and suppression pool temperature.

(6) Support

Support capability e.g., onsite power source (AC & DC) and their associated distribution systems to provid for the shutdown equipment.

7. BWR Equipment Generally Necessary For Cold Shutdown*

At this point the equipment necessary for hot shutdown has reduced th primary system pressure and temperature to where the RHR system may be placed in service in RHR cooling mode.

(1) Decay Heat Removal

Residual heat removal system in the RHR cooling mode, service water system.

(2) Support

Onsite sources (AC & DC) or offsite after 72 hours and their associated distribution systems to provide for shutdown equipment.

* Equipment provided in addition to that for achieving hot shutdown.

8. Information Required For Staff Review

(a) Description of the systems or portions thereof used to provide the shutdown capability and modifications required to achieve the alternate shutdown capability if required.

(b) System design by drawings which show normal and alternate shutdown control and power circuits, location of components, and that wiring which is in the area and the wiring which is out of the area that required the alternate system.

(c) Verification that changes to safety systems will not degrade safety systems. (e.g., new isolation switches and control switches should meet design criteria and standards in FSAR for electrical equipment in the system that the switch is to be installed; cabinets that the switches are to be mounted in should also meet the same criteria (FSAR) as other safety related cabinets and panels; to avoid inadvertent isolation from the control room, the isolation switches should be keylocked, or alarmed in the control room if in the "local" or "isolated" position; periodic checks should be made to verify switch is in the proper position for normal operation; and a single transfer switch or other new device should not be a source for a single failure to cause loss of redundant safety systems).

(d) Verification that wiring, including power sources for the control circuit and equipment operation for the alternate shutdown method, is independent of equipment wiring in the area to be avoided.

(e) Verification that alternate shutdown power sources, including all breakers, have isolation devices on control circuits that are routed through the area to be avoided, even if the breaker is to be operated manually.

(f) Verification that licensee procedure(s) have been developed which describe the tasks to be performed to effect the shutdown method. A summary of these procedures should be reviewed by the staff.

(g) Verification that spare fuses are available for control circuits where these fuses may be required in supplying power to control circuits used for the shutdown method and may be blown by the effects of a cable spreading room fire. The spare fuses should be located convenient to the existing fuses. The shutdown procedure should inform the operator to check these fuses.

(h) Verification that the manpower required to perform the shutdown functions using the procedures of (f) as well as to provide fire brigade members to fight the fire is available as required by the fire brigade technical specifications.

(i) Verification that adequate acceptance tests are performed. These should verify that: equipment operates from the local control station when the transfer or isolation switch is placed in the "local" position and that the equipment cannot be operated from the control room; and that equipment operates from the control room but cannot be operated at the local control station when the transfer or isolation switch is in the "remote" position.

(j) Technical Specifications of the surveillance requirements and limiting conditions for operation for that equipment not already covered by existing Tech. Specs. For example, if new isolation and control switches are added to a service water system, the existing Tech. Spec. surveillance requirements on the service water system should add a statement similar to the following:

"Every third pump test should also verify that the pump starts from the alternate shutdown station after moving all service water system isolation switches to the local control position."

(k) Verification that the systems available are adequate to perform the necessary shutdown functions. The functions required should be based on previous analyses, if possible (e.g., in the FSAR), such as a loss of normal a.c. power or shutdown on a Group I isolation (BWR). The equipment required for the alternate capability should be the same or equivalent to that relied on in the above analysis.

(1) Verification that repair procedures for cold shutdown systems are developed and material for repairs is maintained on site.

FIRE PROTECTION REVIEW

In accordance with section 9.5.1, Branch Technical Position ASB 9.5-1, position C.4.a.(1) of NRC Standard Review Plan and section III.G of new Appendix R to 10 CFR Part 50, it is the staff's position that cabling for redundant safe shutdown systems should be separated by walls having a three-hour fire rating or equivalent protection (see section III.G.2 of Appendix R). That is, cabling required for or associated with the primary method of shutdown, should be physically separated by the equivalent of a three-hour rated fire barrier from cabling required for or associated with the redundant or alternate method of shutdown. To assure that redundant shutdown cable systems and all other cable systems that are associated with the shutdown cable systems are separated from each other so that both are not subject to damage from a single fire hazard, we require the following information for each system needed to bring the plant to a safe shutdown.

1. Provide a table that lists all equipment including instrumentation and vital support system equipment required to achieve and maintain hot and/or cold shutdown. For each equipment listed:

   a. Differentiate between equipment required to achieve and maintain hot shutdown and equipment required to achieve and maintain cold shutdown,

   b. Define each equipment's location by fire area,

   c. Define each equipment's redundant counterpart,

d. Identify each equipment's essential cabling (instrumentation, control, and power). For each cable identified: (1) Describe the cable routing (by fire area) from source to termination, and (2) Identify each fire area location where the cables are separated by less than a wall having a three-hour fire rating from cables for any redundant shutdown system, and

e. List any problem areas identified by item 1.d.(2) above that will be corrected in accordance with Section III.G.3 of Appendix R (i.e., alternate or dedicated shutdown capability).

2. Provide a table that lists Class 1E and Non-Class 1E cables that are associated with the essential safe shutdown systems identified in item 1 above. For each cable listed: (* See note on Page 3).

   a. Define the cables' association to the safe shutdown system (common power source, common raceway, separation less than IEEE Standard-384 guidelines, cables for equipment whose spurious operation will adversely affect shutdown systems, etc.),

   b. Describe each associated cable routing (by fire area) from source to termination, and

   c. Identify each location where the associated cables are separated by less than a wall having a three-hour fire rating from cables required for or associated with any redundant shutdown system.

3. Provide one of the following for each of the circuits identified in item 2.c above:

   (a) The results of an analysis that demonstrates that failure caused by open, ground, or hot short of cables will not affect it's associated shutdown system, * Note

   (b) Identify each circuit requiring a solution in accordance with section III.G.3 of Appendix R, or

   (c) Identify each circuit meeting or that will be modified to meet the requirements of section III.G.2 of Appendix R (i.e., three-hour wall, 20 feet of clear space with automatic fire suppression, or one-hour barrier with automatic fire suppression).

4. To assure compliance with GDC 19, we require the following information be provided for the control room. If credit is to be taken for an alternate or dedicated shutdown method for other fire areas (as identified by item 1.e or 3.b above) in accordance with section III.G.3 of new Appendix R to 10 CFR Part 50, the following information will also be required for each of these plant areas.

   a. A table that lists all equipment including instrumentation and vital support system equipment that are required by the primary method of achieving and maintaining hot and/or cold shutdown.

* NOTE

Option 3a is considered to be one method of meeting the requirements of Section II.G.3 Appendix R. If option 3a is selected the information requested in items 2a and 2c above should be provided in general terms and the information requested by 2b need not be provided.

b. A table that lists all equipment including instrumentation and vital support system equipment that are required by the alternate, dedicated, or remote method of achieving and maintaining hot and/or cold shutdown.

c. Identify each alternate shutdown equipment listed in item 4.b above with essential cables (instrumentation, control, and power) that are located in the fire area containing the primary shutdown equipment. For each equipment listed provide one of the following:

(1) Detailed electrical schematic drawings that show the essential cables that are duplicated elsewhere and are electrically isolated from the subject fire areas, or

(2) The results of an analysis that demonstrates that failure (open, ground, or hot short) of each cable identified will not affect the capability to achieve and maintain hot or cold shutdown.

d. Provide a table that lists Class 1E and Non-Class 1E cables that are associated with the alternate, dedicated, or remote method of shutdown. For each item listed, identify each associated cable located in the fire area containing the primary shutdown equipment. For each cable so identified provide the results of an analysis that demonstrates that failure (open, ground, or hot short) of the associated cable will not adversely affect the alternate, dedicated, or remote method of shutdown.

5. The residual heat removal system is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, this interface most likely consists of two redundant and independent motor operated valves with diverse interlocks in accordance with Branch Technical Position ICSB 3. These two motor operated valves and their associated cable may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire-initiated LOCA through the subject high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

a. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

b. Identify each device's essential cabling (power and control) and describe the cable routing (by fire area) from source to termination.

c. Identify each location where the identified cables are separated by less than a wall having a three-hour fire rating from cables for the redundant device.

d. For the areas identified in item 5.c above (if any), provide the bases and justification as to the acceptability of the existing design or any proposed modifications.

ENCLOSURE 5

APPENDIX A TO SRP SECTION 3.8.4

INTERIM CRITERIA FOR SAFETY-RELATED MASONRY WALL EVALUATION

The purpose of this appendix is to provide minimum design considerations and criteria for the review of safety-related masonry walls which will meet the design standards specified in subsection II of this SRP section.

1. General Requirements

   The materials, testing, analysis, design, construction, and inspection related to the design and construction of safety-related concrete masonry walls should conform to the applicable requirements contained in Uniform Building Code - 1979, unless specified otherwise, by the provisions to this criteria.

   The use of other industrial codes, such as ACI-531, ATC-3, or NCMA, is also acceptable. However, when the provisions of these codes are less conservative than the corresponding provisions of these interim criteria, their use should be justified on a case-by-case basis.

   In new construction, no unreinforced masonry walls will be permitted. For operating plants, existing unreinforced walls will be evaluated by the provisions of these criteria. Plants applying for operating licenses which have already built unreinforced masonry walls will be evaluated on a case-by-case basis.

2. Loads and Load Combinations

   The loads and load combinations shall include consideration of normal loads, severe environmental loads, extreme environmental load, and abnormal loads. Specifically, for operating plants, the load combinations provided in the plant's FSAR shall govern. For operating license applications, the following load combinations shall apply (for definition of load terms, see SRP Section 3.8.4, subsection II.3).

   (a) Service Load Conditions

      (1) $D + L$

      (2) $D + L + E$

      (3) $D + L + W$

      If thermal stresses due to T_o and R_o are present, they should be included in the above containment, as follows:

      (1a) $D + L + T_o +$ $_o$

      (1b) $D + L + T_o + R_o + E$

(1c) $D + L + T_o + R_o + W$

Check load combination for controlling condition for maximum 'L' and for no 'L'.

(b) Extreme Environmental, Abnormal, Abnormal/Severe Environmental, and Abnormal/Extreme Enviromental Conditions

(4) $D + L + T_o + R_o + E'$

(5) $D + L + T_o + R_o + W_t$

(6) $D + L + T_a + R_a + 1.5\ P_a$

(7) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$

(8) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) and the corresponding structural acceptance criteria of should be satisfied first without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these loads, local section strength capacities may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

3. Allowable Stresses

Allowable stresses provided in ACI-531-79, as supplemented by the following modifications/exceptions, shall apply.

(a) When wind or seismic loads (OBE) are considered in the loading combinations, no increase in the allowable stresses is permitted.

(b) Use of allowable stresses corresponding to special inspection category shall be substantiated by demonstration of compliance with the inspection requirements of the NRC criteria.

(c) When tension perpendicular to bed joints is used in qualifying the unreinforced masonry walls, the allowable value will be justified by test program or other means pertinent to the plant and loading conditions. For reinforced masonry walls, all the tensile stresses will be resisted by reinforcement.

(d) For load conditions which represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the allowable working stress may be multiplied by the factors shown in the following table:

Type of Stress	Factor
Axial or Flexural Compression[1]	2.5
Bearing	2.5
Reinforcement stress except shear	2.0 but not to exceed 0.9 fy
Shear reinforcement and/or bolts	1.5
Masonry tension parallel to bed joint	1.5
Shear carried by masonry	1.3
Masonry tension perpendicular to bed joint	
for reinforced masonry	0
for unreinforced masonry[2]	1.3

Notes

(1) When anchor bolts are used, design should prevent facial spalling of masonry unit.

(2) See 3(c).

4. Design and Analysis Considerations

(a) The analysis should follow established principles of engineering mechanics and take into account sound engineering practices.

(b) Assumptions and modeling techniques used shall give proper considerations to boundary conditions, cracking of sections, if any, and the dynamic behavior of masonry walls.

(c) Damping values to be used for dynamic analysis shall be those for reinforced concrete given in Regulatory Guide 1.61.

(d) In general, for operating plants, the seismic analysis and Category I structural requirements of FSAR shall apply. For other plants, corresponding SRP requirements shall apply. The seismic analysis shall account for the variations and uncertainties in mass, materials, and other pertinent parameters used.

(e) The analysis should consider both in-plane and out-of-plane loads.

(f) Interstory drift effects should be considered.

(g) In new construction, no unreinforced masonry wall is permitted; also, all grout in concrete masonry walls shall be compared by vibration.

(h) For masonry shear walls, the minimum reinforcement requirements of ACI-531 shall apply.

(i) Special construction (e.g., multiwythe, composite) or other items not covered by the code shall be reviewed on a case-by-case basis for their acceptance.

(j) Licensees or applicants shall submit QA/QC information, if available, for staff review.

In the event QA/QC information is not available, a field survey and a test program reviewed and approved by the staff shall be implemented to ascertain the conformance of masonry construction to design drawings and specificiations (e.g., rebar and grouting).

(k) For masonry walls requiring protection from spalling and scabbing due to accident pipe reaction (Y_r), jet impingement (Y_j), and missile impact (Y_m), the requirements of SRP Section 3.5.3 shall apply. Any deviation from SRP Section 3.5.3 shall be reviewed and approved on a case-by-case basis.

5. Revision of Criteria

The criteria will be revised, as appropriate, based on:

(a) Design review meetings with the selected licensees and their A/Es.

(b) Experience gained during review.

(c) Additional information developed through testing and researches.

6. References

(a) Uniform Building Code - 1979 Edition.

(b) Building Code Requirements for Concrete Masonry Structures ACI-531-79 and Commentary ACI-531R-79.

(c) Tentative Provisions for the Development of Seismic Regulations for Buildings-Applied Technology Council ATC 3-06.

(d) Specification for the Design and Construction of Load-Bearing Concrete Masonry - NCMA August, 1979.

(e) Trojan Nuclear Plant Concrete Masonry Design Criteria Safety Evaluation Report Supplement - November, 1980.

(f) Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

ENCLOSURE 6

Fracture Prevention of Containment Pressure Boundary (GDC-51)

GDC-51 requires that under operating, maintenance, testing and postulated accident conditions, (1) the Ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The Ferritic materials of the containment pressure boundary which are assessed by the staff are those of components such as freestanding containment vessel, equipment hatches, personnel airlocks, primary containment drywell head, heads containment penetration sleeves, proccess pipes, end closure caps and flued heads and penetrating piping systems downstream of penetration process pipes extending to and including the system isolation valves.

The acceptability of these materials within the context of GDC-51 is determined in accordance with the fracture toughness criteria identified for Class 2 materials by the Summer 1977 Addenda to ASME Code Section III.

TMI TASK ACTION ITEM I.G.1 SPECIAL LOW POWER TEST PROGRAM FOR BWR'S

NUREG-0694 requires applicants to "define and commit to a special low power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training".

A low power test program developed by TVA for Sequoyah and consisting of demonstrations of simulated decay heat removal under degraded plant conditions has been approved for PWR applicants. The "degraded conditions" to which PWR's are being subjected include various combinations of natural circulation and reduced saturation margin operations with actual and simulated A/C power losses, steam generator isolations and boration and cooldown.

In view of the fact that natural circulation and reduced saturation margin conditions are routine to BWR operations, the Sequoyah program cannot be used for BWR's in its entirety. It would be possible to use the standby boron system to perform a boron mixing test similar to one of the Sequoyah tests; however, the experience gained would not justify the cleanup problem. We consider one of the PWR tests, a simulated loss of all A/C power to be feasible and should be performed on BWR's. The objective of this test is to familiarize operators with plant response and determine plant limitations in a blackout. To perform such a test a real or simulated source of decay heat is necessary. (In the PWR programs decay heat is simulated either by input of fission heat or coolant pump heat). To use decay heat it will be necessary to defer the test until decay heat is available (as is permitted for one of the PWR tests in which reactivity control would be difficult). If you opt to perform the test with decay heat, you should perform the test during the first fuel cycle and immediately following 7 days of operation at 80% rated power or above. If you opt to use a simulated source of decay heat such as steam from an external source or actual reactor power, you should perform the test during the intitial test program.

In addition to the above, you should also commit to augmented operator training by participation in the pre-op and startup test programs. Guidelines for the latter will be provided by the BWR Owners' Group. The format for your test procedure should be consistent with Regulatory Guide 1.68. The results of the test should be documented as part of the "Start-up Test Report" (see Regulatory Guide 1.16).

The above actions constitute a basis for satisfactory compliance with Item I.G.1.

BWR OWNERS' GROUP

D.B. Waters Chairman

P.O. Box 1551 • Raleigh, North Carolina 27602 • (919) 836-6584

BWROG-8120

February 4, 1981

U. S. Nuclear Regulatory Commission
Division of Licensing
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: D. G. Eisenhut, Director

Gentlemen:

SUBJECT: BWR OWNERS' GROUP EVALUATION OF NUREG-0737 REQUIREMENT I.G.1, TRAINING DURING LOW POWER TESTING

This letter transmits on behalf of the BWR Owners' Group sixty copies of the BWR Owners' Group program for compliance with the subject requirement.

Requirement I.G.1, which is applicable to near-term operating licence (NTOL) facilities, has been reviewed against the present BWR Preoperational and Startup Test Program. A number of areas are identified where increased emphasis on operator training can be beneficial. Although we believe the scope of the present test program is more than adequate, several new tests are identified that are responsive to the subject NRC requirements as discussed with your staff on September 5, 1980. The additional tests are in accordance with submitted safety analysis reports (SAR's); therefore no new analyses are required to support adding these tests.

The result of the above review is the attached generic program developed by General Electric and the participating Owners listed in Appendix F of the attachment. The generic program will be used as a basis for individual submittals. Implementation details are plant dependent, based on the completion status of the preoperational test program, the scope of the present test and training program, and the plants administration procedures.

The submittal of an Owners' Group position developed in response to an NRC requirement does not indicate that the Owners' Group unanimously endorses that position; rather, it indicates that a substantial number of members believe the position is responsive to the NRC requirement and adequately satisfies the requirement. Each member must formally endorse a position so developed and submitted in order for the position to become the member's position.

Please contact R. H. Buchholz (GE), 408-925-5722 if you have any questions concerning the enclosed information.

Yours very truly,

David B. Waters

D. B. Waters, Chairman
BWR Owners' Group

DBW:PWM:na

Enclosure

cc: BWR Owners' Group
R. H. Buchholz (GE)
P. W. Marriott (GE)
D. M. Verrelli (NRC)

BWR OWNERS' GROUP PROGRAM

FOR COMPLIANCE WITH

NUREG-0737 ITEM I.G.1

TRAINING DURING LOW POWER TESTING

TABLE OF CONTENTS

INTRODUCTION

The NRC has identified new requirements for GE BWR plant testing and training. These requirements are applicable to near-term operating license (NTOL) facilities. The following quotes are from the NUREG documents addressing these requirements:

NUREG 0660 May, 1980

TASK I.G. PREOPERATIONAL AND LOW-POWER TESTING

A. OBJECTIVE: Increase the capability of the shift crews to operate facilities in a safe and competent manner by assuring that training for plant evolutions and off-normal events is conducted. Near-term operating license facilities will be required to develop and implement intensified training exercises during low-power testing programs. This may involve the repetition of startup tests on different shifts for training purposes. Based on experiences from the near-term operating license facilities, requirements may be applied to other new facilities or incorporated into the plant drill requirement (Item I.A.2.5). Review comprehensiveness of test programs.

NUREG 0694 June, 1980

I.G.1 TRAINING DURING LOW-POWER TESTING

Define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

The participating members of the GE BWR Owners'Group, Appendix F, and the General Electric Company have reviewed the present BWR Preoperational and Startup Test Programs against the above listed requirements. A number of areas have been identified where increased emphasis on operator training can be beneficial. Additionally, several new tests have been identified that are responsive to the NRC requirements discussed with the Owners Group subcommittee on September 5, 1980. As a result of this review, a generic program has been developed and is described herein. This generic program will be used as a basis for individual submittals.

The test program has been divided into five sections for purposes of this report. They are:

INTRODUCTION - (Cont'd)

I - Preoperational Testing
II - Cold Functional Testing
III - Hot Functional Testing
IV - Startup Testing
V - Additional Training and Testing

The first four sections briefly discuss the present test program and changes made to improve the training benefit. The last section contains new testing proposed to provide meaningful technical information and enhance training.

I. PREOPERATIONAL TESTING

Following completion of construction tests preoperational tests are performed. The purpose of the preoperational test program is to verify that the performance of plant systems meet design and operational requirements. System components are tested, logic checks are performed, and sensor setpoints are verified. The system is then tested as a whole. The preoperational test program serves a two-fold purpose. Primarily, it controls and documents the preoperational test effort. A secondary benefit of the program is that during the test phase, a detailed knowledge of the systems and their performance characteristics will be obtained by the plant operating group.

Preoperational tests are performed on, as a minimum, any system whose operation is safety related. Plant operating personnel will obtain hands on experience for testing of these systems thereby helping to satisfy the training concerns of NUREGS 0660 and 0694. Many system tests will be conducted as part of these preoperational tests which readily lend themselves to operator training. The Integrated ECCS with loss of AC and DC power test is one of the more significant tests performed during the preoperational test phase which significantly supports operator training. Appendix A describes this test.

To enhance the training benefit of this test future Integrated ECCS testing will be scheduled so that each shift will participate in at least one of these tests to obtain training. Operators obtain an appreciation and feel for control room and plant conditions/limitations and will be required to resolve operational problems associated with the loss of emergency battery and diesel generators during a time when emergency equipment is required to operate.

II. COLD FUNCTIONAL TESTING

Cold Functional Tests are performed on a Plant for several reasons. Some of the more important reasons are as follows:

A. Assure that plant systems are available to support fuel loading.

B. Assure that shift personnel have operating experience with plant equipment.

C. Assure that certain plant operating procedures and surveillance procedures have been tried and are usable.

D. Assure that each shift has functioned together to operate the plant systems on an integrated basis.

E. Assure that specified plant equipment has been tested and the plant and personnel are ready for fuel loading.

The Cold Functional Tests are performed using plant procedures and are controlled and documented by use of checklists. The checklist provides a signoff sheet to assure that each shift has received training and experience on specified systems. Typically, a designated shift supervisory person will be responsible to ensure, by signing the checklist, that their shift has performed the operation specified. Typical systems to be included and an example of a typical checklist are found in Appendix B.

Present testing plans will be reviewed and upgraded, as necessary, to obtain documentation and testing scope for the operator training effort.

III. HOT FUNCTIONAL TESTING

Hot Functional Tests are performed to assure that insofar as possible the system, procedures, and personnel are ready for operation at various power levels. This verification is done by operating systems in an integrated fashion at operating temperatures and pressures at the earliest opportunity for meaningful checks.

The Hot Functional Tests cover those areas of the Plant systems which are not tested by the Startup Test Procedures, but where it is felt that additional data over and above the Cold Functional Tests is beneficial.

Typically, the Hot Functional Tests will begin after fuel is loaded when nuclear heat is available. The Startup provides three phases which offer Hot Functional Test opportunities. These phases are listed below:

A. During heatup from ambient and 0 PSIG to rated temperature and pressure.

B. After increase from rated temperature and pressure to 30 percent power.

C. From 30 percent to 100 percent power.

The Hot Functional Tests are not intended to replace any of the startup test procedures, although there are portions which will be conducted simultaneously.

Those systems whose environment does not change during ascension to rated temperature and pressure will not receive additional testing.

Typical examples of tests, checks, and signoffs to be performed on systems are listed in Appendix C.

During the performance of this testing an Operations Supervisor shall cause a review to be performed of the Control Room copy of the procedures manual to ensure that changes are marked in the manual and with the required approvals as specified by the administrative procedures. He will additionally verify that personnel on each shift have been familiarized with the changes to procedures through the use of information acknowledgements.

Testing plans will be reviewed and upgraded, as necessary, to obtain sufficient documentation and testing scope for the operator training effort.

IV. STARTUP TESTING

A typical startup test program is composed of phases characterized by differences in plant and test conditions. Startup tests are comprised of four phases which include fuel loading and subsequent tests.

1. Open Vessel Testing
2. Initial Heatup
3. Power Tests
4. Warranty Tests

Typical tests to be performed during open vessel, reactor heatup and power ascension are summarized in Figures 1 and 2.

The actual testing sequence will be determined at each site. The recommended normal testing sequence can be obtained from Figure 1: Start from the left side of the page and move to the right, completing each column of tests before proceeding to the next column (example - all open vessel tests should be completed before heatup tests are started). The notable exception is that testing at natural circulation on the 100% load line (Test Condition 4) will normally be done following pump trips from Test Condition 6. The normal recommended sequence of tests in a column would be: 1) core performance analysis, 2) steady state testing, 3) control system tuning and 4) major trips. The actual testing sequence can vary from recommended test sequence due to equipment problems and other considerations.

Typical startup tests are described in the brief summaries of Appendix D. These tests were chosen from the tests listed in Figure 1 to provide insight into operator training obtained during this period.

The significant training related, startup tests will be balanced so that each operating shift will:

1. See at least one reactor scram transient.

2. See at least one pressure regulator transient.

3. See at least one turbine trip transient or load rejection.

4. Operate the RCIC (and if applicable HPCI) system.

5. See at least one water level setpoint transient.

The other testing will be balanced as much as practicable to ensure even exposure to testing for all operating shifts.

V. ADDITIONAL TRAINING AND TESTING

Upgrading the training program for the presently defined test program will meet the training and testing intent of the NUREG sections quoted in the INTRODUCTION section of this report. However, in response to information obtained at the 9/5/80 meeting held with the NRC and because of our efforts to provide as comprehensive a test program as possible several new tests will be added to the test program. These tests will provide additional technical information to aid in system and plant operational readiness evaluations. The tests will also provide some operator training. These tests will be performed once per plant and significant training information obtained will be transmitted to non-participating personnel via test critiques.

Appendix E contains test descriptions defining the scope of the tests to be added to the test programs. Each facility will write Detailed procedures that will be prepared for individual plants within the scope of those descriptions.

CONCLUSIONS

As explained in this report, each phase of the testing program provides a building block for the next phase and provides the necessary overlap and depth to ensure that the facility's operating staff will obtain maximum meaningful inplant training to assure that crews will operate their facilities in a safe and competent manner and that all safety related systems are thoroughly tested. We are confident that, as delineated in this report, the increased emphasis on operator training and the addition of new testing, when coupled with the present testing and training programs, more than adequately satisfies the requirements of I.G.1 of NUREGs 0660 and 0694.

APPENDIX A

EVENT: INTEGRATED ECCS WITH LOSS OF AC & DC POWER TEST

The Integrated ECCS Test is performed to demonstrate the following:

A. If applicable, the capability of the startup transformer with interconnected buses and the station battery systems with interconnected buses to start all the core standby cooling systems.

B. The response of the diesel generators and interconnected equipment to a loss of off-site power (no loss of coolant).

C. The capability of the diesel generators with the load shedding logic to auto start and assume all their respective emergency core cooling loads under a loss of offsite power, loss of coolant accident signal (LOCA).

D. The capability of the above systems to provide sufficient emergency core cooling equipment during LOCA conditions with "A" DC bus and associated emergency AC bus deenergized.

E. The capability of the above system to provide sufficient emergency core cooling equipment during LOCA conditions with "B" DC bus and associated AC bus deenergized.

F. The capability of the above systems to provide sufficient emergency core cooling equipment during LOCA conditions with each remaining individual emergency DC and associated emergency AC bus deenergized.

G. These tests are run for a sufficiently long period of time to verify proper separation between emergency power systems.

Typically, the following tests are performed:

1. Simulated LOCA (with offsite power available).

2. Loss of offsite power (LOSP) with simultaneous simulated LOCA.

3. LOSP with simultaneous simulated LOCA coincident with a loss of the "A" emergency DC battery system and associated emergency AC diesel generator.

4. LOSP with simultaneous simulated LOCA coincident with a loss of the "B" emergency DC battery system and associated emergency AC diesel generator.

5. Test 4 is repeated substituting each remaining emergency DC and associated emergency AC diesel generator for the "B" system until all systems are tested.

APPENDIX B

Typical systems to be included as part of this program are:

Main Steam Systems

Main Steam Isolation Valves
Main Steam Relief Valves
Turbine Seal and Steam Air Ejectors

Reactor Vessel & Auxiliary Systems

Recirculation System
Reactor Water Cleanup System
Control Rod Drive System
Reactor Vessel Level Instrumentation
Standby Liquid Control
Remote Shutdown System

ECCS System

LPCS
RHR (including LPCI, Shutdown Cooling, Suppression Pool Cooling and Suppression Pool Spray Modes)
HPCI (if applicable)
HPCS (if applicable)

Emergency Electrical System

Diesel Generator, and Emergency Buses
Emergency Batteries
Vital AC System

Plant Support Systems

Service Water
Reactor Building Closed Cooling Water
Turbine Building Closed Cooling Water
Radwaste
Makeup Demineralizer
Fuel Pool Cooling
Demineralized Water Transfer and Storage

APPENDIX B

Plant Support Systems (cont'd.)

Condensate Transfer and Storage

Instrument and Service Air

Ventilation

Emergency Service Water

Circulating Water

SYSTEM TRAINING - PROCEDURE AND EXPERIENCE CHECKS

SYSTEM ______________________________

1) SHIFT FOREMAN HAS CONDUCTED A REVIEW OF THE NORMAL OPERATING PROCEDURE WITH THE SHIFT PERSONNEL.
PROCEDURE NO. ______________________________

2) THE SHIFT PERSONNEL HAVE OPERATED THE SYSTEM AS SPECIFIED BELOW:

____

____

____

3) THE SHIFT FOREMAN HAS CONDUCTED A REVIEW OF THE FOLLOWING EMERGENCY OPERATING PROCEDURES:

PROCEDURE NO. ______________________________

PROCEDURE NO. ______________________________

PROCEDURE NO. ______________________________

PROCEDURE NO. ______________________________

4) THE SHIFT FOREMAN HAS CONDUCTED ORAL EXAMINATION OF HIS SHIFT PERSONNEL CONCERNING THE SYSTEM AND, IN HIS JUDGEMENT, THE PERSONNEL HAVE ADEQUATE KNOWLEDGE OF SYSTEM OPERATION.

5) SIGN OFF OF ITEMS 1, 2, 3, AND 4.

DAY SHIFT ______________________________
SF DATE

EVENING SHIFT ______________________________
SF DATE

MIDNIGHT SHIFT ______________________________
SF DATE

APPENDIX C

DURING HEATUP FROM AMBIENT AND 0 PSIG TO RATED TEMPERATURE AND PRESSURE

SYSTEM	MODE OF OPERATION AND HOT FUNCTIONAL TESTS
CRD System	In continuous normal operation, check each fully withdrawn CRD for coupling as it is withdrawn. Observe temperatures are in limits. Observe for proper position indication. Record rod patterns.
Drywell Leakage Detection System	Monitor sump pump integrators which should be in continuous operation. Determine identified and unidentified leakage rates at 500 and 920 PSIG.
Drywell Temp. and Drywell Cooling	Both should be in continuous operation per procedure.
Process Radiation Monitors	In continuous operation. Check for response to increasing power levels.
Ventilation System	In continuous operation. Check that steam tunnel temperature is within temperature limits at rated temperature and pressure. Verify proper operation of leakage detection systems.
Turbine EHC Pressure Controls	Start heatup with controlling regulator set at 150 PSIG and by-pass opening jack at 0. When reactor pressure is greater than 150 PSIG check that regulator responds to setpoint changes.
Rod Worth Minimizer	In continuous operation. Verify proper operation as rods are withdrawn.
Main Steam Relief Valves	Record the discharge throat TC and pressure readings from recorder and determine that the valves do not have seat leakage.
Condensate Demineralizer System	Verify performance of system to adequately control water quality by observing that water quality stays within limits specified by plant chemist. Check (if applicable) demineralizer bypass valves not in auto.

APPENDIX C

SYSTEM	MODE OF OPERATION AND HOT FUNCTIONAL TESTS
TIP System	Make trail traces if flux level permits. Verify leak tightness and air/nitrogen purge.
Reactor Water Cleanup System	In continuous operation at approximately 50 percent to 100 percent flow. Place cleanup recirculation pumps in operation at pressure and operate in all modes. Check that valves operate properly. Reject reactor water back to condenser and radwaste to check reject valve for proper operation.
Reactor Recirculation System	In continuous operation per operating procedure. Check that seal cavity, oil reservoir, winding temperatures, and MG set temperatures are within limits. Check that cavity pressures follows heatup pressure. Check that recirc. loop temperature recorder indicates the proper temperature increase.
Condensate and Feedwater	In continuous operation to maintain reactor level. Start standby feed pump turbine per procedure, place in service and remove replaced turbine from service.
SRM and IRM	In continuous operation. Check for proper retract operation as they are withdrawn. Insert and check for proper operation/indication.
Turbine Seal	Place in continuous operation per operating procedure. Check that seal steam regulator controls seal pressure. Place backup regulator in service.
Vacuum Pump	Place in service per operating procedure.
Steam Jet Air Ejectors	Place in service per operating procedure. Place backup air ejectors in service.

APPENDIX C

SYSTEM	MODE OF OPERATION AND HOT FUNCTIONAL TESTS
Reactor Vessel Temps and Head Leak Detection	Should be in continuous service. Temperatures should be controlled such that vessel temperature differentials are within limits. Head seal leak detector should be valved per operating procedure and observed for seal leakage.
Circulating Water	Continuous operation to maintain adequate condenser vacuum. Shift modes of system operation.

AFTER INCREASE FROM RATED

TEMPERATURE AND PRESSURE TO 30% POWER

A few significant system environmental changes occur between arrival at rated temperature and pressure and completion of 30 percent testing which requires the following additional hot functional checks.

SYSTEM	MODE OF OPERATION AND HOT FUNCTIONAL CHECKS
Turbine Generator	During this period the turbine generator will be placed in operation for the first time and the following checks should be performed which are not part of the formal test program. Verify procedure for turbine warmup and roll to 1,800 RPM. Perform the turbine generator no-load tests. Check turbine vibration at critical speed and 1,800 RPM okay. Verify proper operation of stator cooling and generator seal oil systems. Verify operator familiarization with turbine generator instrumentation and controls both local and remote. Verify oil flow indication at each bearing inspection spout. Verify that expansion (stretchout) is satisfactory. Perform over-speed checks.

APPENDIX C

SYSTEM	MODE OF OPERATION AND HOT FUNCTIONAL CHECKS
Feedwater Heater Controls	Put feedwater heaters in service, and establish level control. Feedwater temperature will rise. Inspect feedwater line and feedwater pump casings to assure thermal expansion has not opened flanges or affected mechanical seal operation.
RBCCW System	Check temperatures of cooled components. Readjust as necessary to maintain proper temperature in components as specified in the operating procedures.

DURING OPERATION FROM 30 PERCENT TO 100 PERCENT

POWER

At this point, all safety-related equipment and procedures have been checked out by the combination of cold functional tests, surveillance tests, hot functional tests, and the startup tests, performed thus far. The startup test program adequately tests remaining plant performance and operating procedures associated with delivering greater than 30 percent power to network.

The following is an example of the format for a Hot Functional test signoff:

Control Rod Drive System	Shift Foreman Operations Supervisor/INITIALS
1. Checks required are complete.	____________ / ______
2. System performance adequate to proceed.	____________ / ______
3. Operating procedures modified if necessary.	____________ / ______
4. All shifts knowledgeable of system operation and procedure changes.	____________ / ______

APPENDIX D

RCIC System

Purpose

The purpose of this test is to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) system over its expected operating pressure range.

Description

The RCIC system test consists of two parts: injection to the condensate storage tank and injection to the reactor vessel. The CST injections consist of controlled and quick starts at reactor pressures ranging from 150 psig (10.5 kg/cm^2) to rated, with corresponding pump discharge pressures throttled between 100 psig (17.6 kg/cm^2) and 250 psig above rated pressure. During this part of the testing, proper operation of the system will be verified and adjustments made as required to meet this criteria. A cold quick start and two hours of continuous operation will be demonstrated. The cold quick start requires a minimum of three days with no RCIC operation. The reactor vessel injection will consist of a cold quick start of the system with all flow routed to the reactor vessel at 25% power.

PRESSURE REGULATOR

Purpose

The purposes of this test are a) to determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators, b) to demonstrate the takeover capability of the backup pressure regulator via simulated failure of the controlling pressure regulator and to set the regulating pressure difference between the two regulators to an appropriate value c) to demonstrate smooth pressure control transition between the turbine control valves and bypass valves when the reactor steam generation exceeds the steam flow used by the turbine.

APPENDIX D

PRESSURE REGULATOR

Description

The pressure setpoint will be decreased and then increased rapidly by about 10 psi (0.7 kg/cm^2) and the response of the system will be measured in each case. It is desirable to accomplish the setpoint change in less than 1 second. At specified test conditions the load limit setpoint will be set so that the transient is handled by control valves, bypass valves and both. The back-up regulator will be tested by simulating a failure of the operating pressure regulator so that the back-up regulator takes over control. The response of the system will be measured and evaluated and regulator settings will be optimized.

FEEDWATER SYSTEM

Purpose

The purposes of this test are a) to adjust the feedwater control system for acceptable reactor water level control, b) to demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump, c) to demonstrate adequate response to feedwater temperature loss, and d) to determine the maximum feedwater runout capability.

Description

Reactor water level setpoint changes of approximately 5 to 6 inches (12.5 to 15.3 cm) will be used to evaluate and adjust the feedwater control system settings for all power and feedwater pump modes. The level setpoint changes will also demonstrate core stability to subcooling changes. One of two operating feedwater pumps will then be tripped and the automatic flow runback circuit will act to drop power to within th capacity of the remaining pump. The worst single failure case of feedwater temperature loss will be performed and the resulting transients recorded between 80 and 90% power and near rated core flow rate. Data will be taken between 50 and 100% power to allow the determination of the maximum feedwater runout capability.

APPENDIX D

MAIN STEAM ISOLATION VALVES

Purpose

The purposes of this test are a) to functionally check the main steam line isolation valves (MSIVs) for proper operation at selected power levels, b) to determine isolation valve closure times c) to determine maximum power at which full closures of a single valve can be performed without a scram and d) to determine the reactor transient behavior resulting from the simultaneous full closure of all MSIVs.

Description

At 5% and greater reactor power levels, individual fast closure of each MSIV will be performed to verify their functional performance and to determine closure times. The MSIV closure times will be determined from the MSL isolation data.

To determine the maximum power level at which full individual closures can be performed without a scram actuation will be performed between 50 and 65% power and used to extrapolate to the next test point between 70 and 85% power, and ultimately to the maximum power test condition with ample margin to scram.

A test of the simultaneous full closure of all MSIVs will be performed at >75% of rated thermal power. Correct performance of the RCIC and relief valves will be shown. Reactor process variables will be monitored to determine the transient behavior of the system during and following the Main Steam Line (MSL) isolation.

TURBINE TRIP AND GENERATOR LOAD REJECTION

Purpose

The purpose of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and the generator.

APPENDIX D

TURBINE TRIP AND GENERATOR LOAD REJECTION

Description

Turbine Trip (closure of the main turbine stop valves within approx. 0.1 second) and Generator Trip (closure of the main turbine control valves in about 0.1 to 0.2 seconds) will be performed at selected power levels during the Startup Test Program. At low power levels, reactor protection following the trip is provided by high neutron flux and vessel high pressure scrams. For the protective trips occurring at intermediate and higher power levels, reactor will scram by relays, actuated by stop/control valve motion.

A generator trip will be performed at low power level such that nuclear boiler steam generation is just within the bypass valve capacity to demonstrate scram avoidance.

For the trips performed at intermediate power range, reactor scram is most important in controlling the transient peaks.

Above about 40% power, the recirculation pump circuit breakers are both automatically tripped (RPT) and subsequent transient pressure rise will be limited by the opening of the bypass valves initially, and the safety relief valves, if necessary.

For the turbine trip, the main generator breakers remain closed for a time so there is no rise in turbine generator speed, whereas, in the generator trip, the main generator breaker opens and the residual turbine steam will cause a momentary rise in the generator speed.

SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

Purpose

The purpose of this test is to demonstrate that the reactor can be brought from a normal initial steady-state power level to the point where cooldown is initiated and under control with reactor vessel pressure and water level controlled from outside the control room.

APPENDIX D

SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

Description

The test will simulate the reactor shutdown following a control room evacuation. The reactor will be scrammed from a normal steady-state condition, the vessel water level and pressure will be controlled from outside the control room. All other operator actions not directly related to vessel water level and pressure will be performed in the main control room.

RECIRCULATION SYSTEM (for variable speed MG set plants)

Purpose

The purposes of this test are 1) to obtain recirculation system performance data under different operational conditions, such as pump trip, flow coastdown, pump restart 2) to verify that no recirculation system cavitation will occur in the operable region of the power-flow map and 3) to verify that, during the trip of recirculation pumps, the feedwater control system can satisfactorily control water level without a resulting turbine trip/scram, and to record and verify acceptable performance of the recirculation pump circuit breaker trip system (RPT).

Description

Recirculation pump trips are performed to determine reactor water level transient characteristics. The reactor transient response during the trip and coastdown of one recirculation M-G set and its pump will be determined. All single-pump trips will be initiated by tripping either the M-G Set drive motor breaker or field breaker. Single pump trips of one M-G set drive motor will be used to determine the Feedwater Control System Transient performance. These transients will be extrapolated to field breaker trip of one pump. In case of higher power turbine or generator trips, there is an automatic opening of circuit breakers in the pump power supply. The result is a fast core flow coastdown that helps reduce peak neutron and heat flux in such events. The two pump circuit breaker trip at test condition 3 provides the best opportunity to observe the drive flow and core flow coastdowns while not being greatly affected by other transients, as in the midst of a T/G trip.

APPENDIX D

LOSS OF TURBINE-GENERATOR AND OFFSITE POWER

Purpose

This test determines electrical equipment and reactor system transient performance during a loss of auxiliary power.

Description

The Loss of Auxiliary Power Test will be performed at 20% to 30% of rated power. The proper response of the reactor plant equipment, automatic switching equipment and the proper sequencing of the diesel generator loads will be checked. Appropriate reactor parameters will be recorded during the resultant transient.

APPENDIX E

TEST: STARTUP OF THE RCIC SYSTEM AFTER LOSS OF AC POWER TO THE SYSTEM.

PURPOSE: VERIFY THE DESIGN BASIS ABILITY OF THE SYSTEM TO START WITHOUT THE AID OF AC POWER WITH THE EXCEPTION OF THE RCIC DC/AC INVERTERS.

INITIAL CONDITIONS:

- PREOPERATIONAL TEST HAS BEEN PERFORMED ON RCIC SYSTEM.
- IF TEST IS PERFORMED PRIOR TO THE AVAILABILITY OF NUCLEAR STEAM, SUFFICIENT AUXILIARY BOILER CAPACITY AND PIPING TO RUN THE RCIC TURBINE/PUMP MUST BE AVAILABLE.
- SYSTEM VALVES IN NORMAL STANDBY LINEUP (SUCTION FROM CST)

  NOTE: 1) IF THE AUXILIARY BOILER IS USED AS THE TURBINE STEAM SUPPLY, TAG CLOSED THE DRYWELL STEAM SUPPLY ISOLATION VALVE.

  2) FLOW CAN EITHER BE DIRECTED TO THE REACTOR PRESSURE VESSEL OR BACK TO THE CONDENSATE STORAGE TANK.
- POWER TO ALL RCIC COMPONENTS FED BY SITE AC POWER SHALL BE SECURED.
- STATION BATTERIES SHALL BE FULLY CHARGED.
- INSTRUMENT AIR SHALL BE AVAILABLE FOR OPERATION AND CONTROL OF APPLICABLE VALVES.
- INSTRUMENTS SHALL BE CALIBRATED AND SETPOINTS, WHERE APPLICABLE, SHALL BE VERIFIED.

TEST DESCRIPTION:

PERFORM A MANUAL INITIATION OF THE RCIC SYSTEM UTILIZING THE MANUAL INITIATION SWITCH AND VERIFY THE PROPER OPERATION OF ALL COMPONENTS REQUIRED FOR THE RCIC STARTUP TRANSIENT TO RATED FLOW.

NOTE: MANUAL MANIPULATION OF SOME VALVES WILL BE REQUIRED IF FLOW IS RETURNED TO THE CST OR AUXILIARY BOILER STEAM IS USED.

ACCEPTANCE CRITERIA:

PROPER OPERATION OF ALL COMPONENTS FOR THE RCIC STARTUP TRANSIENT UNTIL RATED FLOW IS OBTAINED.

APPENDIX E

TEST: OPERATION OF THE RCIC SYSTEM WITH A SUSTAINED LOSS OF AC POWER TO THE SYSTEM

PURPOSE: TO VERIFY THE OPERATION OF RCIC BEYOND ITS DESIGN BASIS TO EVALUATE THE LIMITS OF SYSTEM OPERATION WITH EXTENDED LOSS OF AC POWR TO IT AND SUPPORT SYSTEMS WITH THE EXCEPTION OF THE RCIC DC/AC INVERTERS.

INITIAL CONDITIONS:

- PREOPERATIONAL TEST HAS BEEN PERFORMED ON RCIC SYSTEM.
- IF TEST IS PERFORMED PRIOR TO THE AVAILABILITY OF NUCLEAR STEAM, SUFFICIENT AUXILIARY/BOILER CAPACITY AND PIPING TO RUN THE RCIC TURBINE/PUMP MUST BE AVAILABLE.
- SYSTEM VALVES IN NORMAL STANDBY LINEUP (SUCTION FROM CST).

  NOTE: 1) THE AUXILIARY BOILER IS USED AS THE TURBINE STEAM SUPPLY, TAG CLOSED THE DRYWELL STEAM SUPPLY ISOLATION VALVE.
- POWER TO ALL RCIC COMPONENTS FED BY SITE AC POWER SHALL BE SECURED, INCLUDING RCIC AREA COOLERS AND BATTERY CHARGERS SUPPLYING THE STATION BATTERY FROM WHICH RCIC DC LOADS ARE POWERED.
- RCIC BATTERIES SHALL BE FULLY CHARGED.
- INSTRUMENT AIR SHALL BE AVAILABLE FOR OPERATION AND CONTROL OF APPLICABLE VALVES.
- INSTRUMENTS SHALL BE CALIBRATED AND SETPOINTS, WHERE APPLICABLE, SHALL BE VERIFIED.

TEST DESCRIPTION:

START AND OPERATE THE RCIC SYSTEM WITH RETURN TO THE CST AND RUN FOR 2 HOURS OR UNTIL ANY SYSTEM LIMITING PARAMETER IS APPROACHED (E.G., HIGH RCIC AREA TEMP, LOW BATTERY VOLTAGE, OR HIGH SUPP. POOL TEMP) TRIPPING AND RESTARTING THE RCIC SYSTEM TWO ADDITIONAL TIMES DURING THIS OPERATING PERIOD.

ACCEPTANCE CRITERIA:

NONE

APPENDIX E

TEST: RCIC OPERATION TO PROVE DC SEPARATION.

PURPOSE: VERIFY PROPER OPERATION OF THE RCIC DC COMPONENTS WHEN NON RCIC STATION BATTERIES ARE DISCONNECTED.

INITIAL CONDITIONS:

- PREOPERATIONAL TEST HAS BEEN PERFORMED ON RCIC SYSTEM.
- TEST TO BE PERFORMED PRIOR TO FUEL LOAD.
- THIS TEST IS PERFORMED PRIOR TO THE AVAILABILITY OF NUCLEAR STEAM, SUFFICIENT AUXILIARY BOILER CAPACITY AND PIPING TO RUN THE RCIC TURBINE/PUMP MUST BE AVAILABLE.
- SYSTEM VALVES IN NORMAL STANDBY LINEUP (SUCTION FROM CST).
- DRYWELL STEAM SUPPLY ISOLATION VALVE TAGGED SHUT.
- STATION BATTERIES SHALL BE FULLY CHARGED.
- INSTRUMENT AIR SHALL BE AVAILABLE FOR OPERATION AND CONTROL OF APPLICABLE VALVES.
- INSTRUMENTS BE CALIBRATED AND SETPOINTS, WHERE APPLICABLE, BE VERIFIED.

TEST DESCRIPTION:

START AND OPERATE THE RCIC SYSTEM WITH RETURN TO THE CST. DURING SYSTEM OPERATION DISCONNECT, EACH NON-RCIC STATION BATTERY FROM ITS BUS AND VERIFY PROPER OPERATION OF EACH RCIC DC COMPONENT.

ACCEPTANCE CRITERIA:

PROPER OPERATION OF RCIC DC COMPONENTS WITH NON-RCIC STATION BATTERIES DISCONNECTED.

APPENDIX E

TEST: INTEGRATED REACTOR PRESSURE VESSEL LEVEL FUNCTIONAL TEST.

PURPOSE: VERIFY THAT INSTRUMENTS CONNECTED TO THE RPV ARE TUBED PROPERLY, THAT THE TUBING IS NOT BLOCKED AND THAT INSTRUMENT TRACKING IS PROPER.

INITIAL CONDITIONS:

- ALL INSTRUMENTS CONNECTED TO THE RPV HAVE BEEN CALIBRATED AND ARE OPERABLE.
- RPV HAS BEEN FLUSHED AND IS CLEAN.
- ALL RPV INSTRUMENT TUBING HAS BEEN FILLED, ALL INSTRUMENTS ARE VENTED, AND PROPER VALVE LINEUP VERIFIED.
- A SOURCE OF DEMINERALIZED WATER IS AVAILABLE TO FILL THE RPV.
- FUEL HAS NOT BEEN LOADED INTO THE RPV.
- RPV HEAD REMOVED OR ADEQUATELY VENTED TO PREVENT PRESSURIZATION.

TEST DESCRIPTION:

RAISE AND LOWER (OR LOWER AND RAISE, WHICHEVER IS MOST CONVENIENT) THE RPV WATER LEVEL THROUGH THE RANGE OF RPV LEVELS NECESSARY TO VERIFY THE PROPER OPERATION AND TRACKING OF EACH RPV CONNECTED INSTRUMENT.

NOTE: THE TEMPERATURE AND PRESSURE CONDITIONS AT WHICH THIS TEST WILL BE PERFORMED ARE NOT THE CONDITIONS FOR WHICH THE VARIOUS INSTRUMENTS ARE CALIBRATED. THERE WILL NOT BE A ONE-TO-ONE CORRESPONDENCE BETWEEN ACTUAL REACTOR VESSEL LEVEL CHANGE AND INDICATED LEVEL CHANGE.

ACCEPTANCE CRITERIA:

EACH AFFECTED RPV INSTRUMENTS OPERATION AND TRACKING IS SATISFACTORY.

APPENDIX E

TEST: INTEGRATED CONTAINMENT PRESSURE INSTRUMENTATION TEST (TEST TO BE PERFORMED IN CONJUNCTION WITH CONTAINMENT INTEGRATED LEAK RATE TESTING)

PURPOSE: VERIFY THE PROPER CONNECTION, AND TRACKING OF CONTAINMENT PRESSURE INSTRUMENTS AND THAT THE TUBING SUPPLYING THESE INSTRUMENTS IS NOT BLOCKED.

INITIAL CONDITIONS:

- ALL INITIAL CONDITIONS FOR CONTAINMENT INTEGRATED LEAK RATE TESTING HAVE BEEN ESTABLISHED.
- ALL CONTAINMENT PRESSURE INSTRUMENTS HAVE BEEN CALIBRATED AND ARE VALVED INTO SERVICE.

TEST DESCRIPTION:

AS CONTAINMENT PRESSURE IS INCREASED, DURING THE CONTAINMENT INTEGRATED LEAK RATE TEST, VERIFY PROPER TRACKING OF ALL CONTAINMENT PRESSURE INSTRUMENTS.

ACCEPTANCE CRITERIA:

ALL CONTAINMENT INSTRUMENTS TRACK PROPERLY AND ALL AFFECTED INSTRUMENT LINES ARE CLEAR OF OBSTRUCTIONS.

FIGURE 1

STARTUP TEST SPECIFICATIONS

STI NO.	TEST NAME	OPEN VESSEL	HEAT UP	TEST CONDITIONS[7] 1	2	3	4	5	6		WARRANTY
1	Chemical & Radiochemical	X	X	X		X		X	X		
2	Radiation Measurements	X	X	X		X			X		
3	Fuel Loading	X									
4	Full Core Shutdown Margin	X									
5	CRD	X	X		X	X			X		
6	SRM Perf. & Control Rod Seq.	X	X		X						
8	Rod Sequence Exchange				X						
9	Water Level Measurements		X	X	X	X	X	X	X		
10	IRM Performance	X	X	X							
11	LPRM Calibration			X		X			X		
12	APRM Calibration		X	X	X	X		X	X		X
13	Process Computer	X	X	X[3]					X		
14	RCIC		X		X						
15	HPCI		X			X					
16	Selected Process Temperatures		X			X			X		
17	System Expansion		X	X	X	X					
18	Core Power Distribution					X			X		
19	Core Performance			X	X	X	X	X	X		X
20	Steam Production										X
21	Core Power Void Mode Response						X	X			
22	Pressure Regulators: Setpoint Changes			X, BP	X	X, No BP	X	X	X, A[13]		
	Backup Regulator			X, BP	X	X, No BP	X	X	X, A[13]		
23	FW System: FW Pump Trip								M[12]		
	Water Level Setpoint Change				X	X	X	X	X, A		
	Heater Loss								X[10]		
24	Turbine Valve Surveillance				X[4]			X[5], SP	X[5,6], SP		
25	MSIVs: Each Valve		X	X[3], SP							
	One Valve				X[4], SP			X[5,6], SP			
	Full Isolation								X[2,6,9], SD		
26	Relief Valves: Flow Demonstration				X[4,11]						
	Operational		X[11]		X[4]						
27	Turbine Stop Valve Trip and					X[2,11], SD			X[2,11] SD*		
	Generator Load Rejection				L, SP				*		
28	Shutdown From Outside C Room				X, SD						
29	Recirculation Flow Control System		X	X	M[4], A[4]	M, A[12]		M[5], A[5]	M, A		
30	Recirc. Sys: Trip One Pump					X[11]			X[11]		
	Trip Two Pumps					X[11]					
	System Performance				X	X	X		X		
	Non-Cavit. Verif.					X					
31	Loss of T-G, Offsite Power				X[2], SD						
33	Drywell Piping Vibration				X	X			X		
35	Recirc. System Flow Calibration	X				X			X		
70	Reactor Water Cleanup System		X								
71	Residual Heat Removal System		X[11]	X							

Figure 1. Startup Test Program

1 See Figure 2 for Test Conditions region map.
2 Perform Test 5, timing of 4 slowest control rods in conjunction with these scrams.
3 Between Test Conditions 1 and 3.
4 Between Test Conditions 2 and 3.
5 Between Test Conditions 5 and 6.
6 Before 100% Turbine Trip.
7 Not Applicable
8 Determine maximum power without scram.
9 Anywhere > 75% Power.
10 80-90% Power.
11 Do STI 33 in conjunction with this test.
12 Demonstrate Recirculation System's Runback Feature.
13 Down setpoint only.

L = Local Flow Control Mode
M = Master Manual Flow Control Mode
N = Local or Master Manual Flow Control Mode
A = Automatic Flow Control Mode
SP = Scram Possibility
SE = Scram Expected
SD = Scram Definite
BP = Bypass Valve Response
* = Do either Stop Valve or Control Valve Trip.

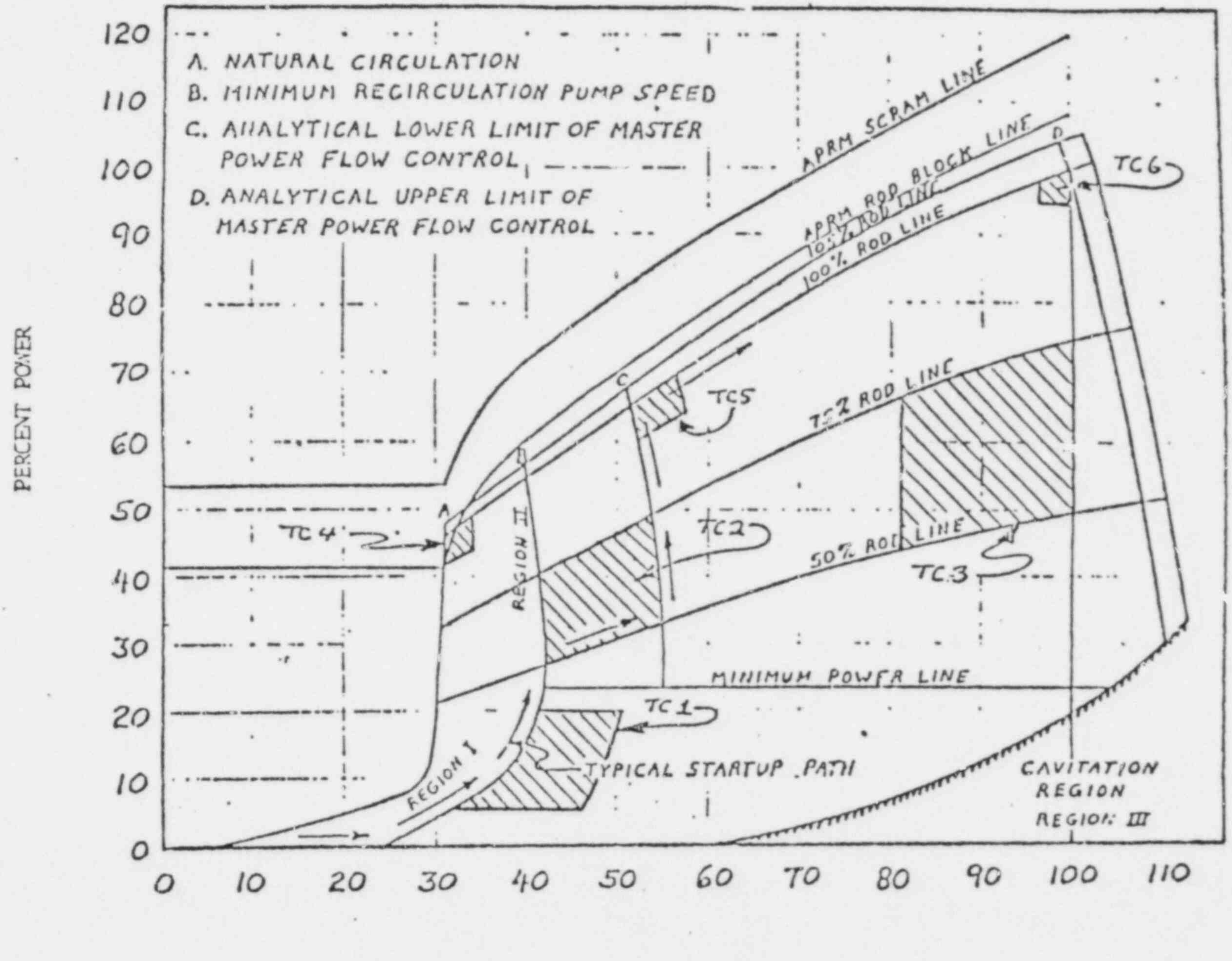

FIGURE 2

SHEET 1 of 2

FIGURE 2

DEFINITION OF TEST CONDITION REGIONS

Test Condition No.	Power-Flow Map Region and Notes
1	Before main generator synchronization-between 5% and 20% thermal power-within 10% of M-G Set minimum operating speed line in Local Manual mode.
2	After main generator synchronization-between the 50% and 75% power rod lines-between M-G Set minimum speeds for Local Manual and Master Manual modes the lower power corner must be less than Bypass Valve capacity.
3	Between the 50% and the 75% control rod lines, with core flow rated between 80% and 100% of its rated value.
4	On the natural circulation core flow line - within 5% of the intersection with the 100% power rod line.
5M	Within 5% of the 100% power rod line - within + 5% of the minimum M-G Set speed for Master Manual mode - Recirculation System engaged in Master Manual mode only.
5A	Within 5% of the 100% power rod line - within + 5% of the core flow rate at the lower end of the Auto Flow Control region - Recirculation System Engaged in Auto Flow Control mode only.
6	Between 95% and 100% of rated power and between 95% and 100% of rated core flow rate.

APPENDIX F

NUREG-0737 ITEM I.G.1

This report applies to the following plants, whose Owners participated in the report's development.

Detroit Edison	Enrico Fermi 2
Long Island Lighting	Shoreham
Mississippi Power & Light	Grand Gulf 1 & 2
Pennsylvania Power & Light	Susquehanna 1 & 2

PRESERVICE AND INSERVICE INSPECTIONS

121.0 MATERIALS ENGINEERING BRANCH

We require that your inspection program for Class 1, 2 and 3 components be in accordance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g). Accordingly, submit the following information:

(1) A preservice inspection plan which is consistent with the required edition of the ASME Code. This inspection plan should include any exceptions you propose to the Code requirements.

(2) An inservice inspection plan submitted within six months of the anticipated date for commercial operation.

This preservice inspection plan will be required to support the safety evaluation report finding regarding your compliance with preservice and inservice inspection requirements. Our determination of your compliance will be based on the edition of Section XI of the ASME Code referenced in your FSAR or later editions of Section XI referenced in the FEDERAL REGISTER that you may elect to apply.

Your response to this item should define the applicable edition(s) and subsections of Section XI of the ASME Code. If any of the examination requirements of the particular edition of Section XI you referenced in the FSAR cannot be met, a request for relief must be submitted, including complete technical justification to support your request.

Detailed guidelines for the preparation and content of the inspection programs to be submitted for staff review and for relief requests are attached as an Appendix to Section 121.0 of our review questions.

APPENDIX TO SECTION 121.0

GUIDANCE FOR PREPARING PRESERVICE AND INSERVICE INSPECTION PROGRAMS AND RELIEF REQUESTS PURSUANT TO 10 CFR 50.55a(g)

A. Description of the Preservice/Inservice Inspection Program

This program should cover the requirements set forth in Section 50.55a(b) and (g) of 10 CFR Part 50; the ASME Boiler and Pressure Vessel Code, Section XI Subsections IAW, IWB, IWC and IWD; and Standard Review Plans 5.2.4 and 6.6. The guidance provided in this enclosure is intended to illustrate the type and extent of information that should be provided for NRC review. It also describes the information necessary for "request for relief" of items that cannot be fully inspected to the requirements of Section XI of the ASME Code. By utilizing these guidelines, applicants can significantly reduce the need for requests for additional information from the NRC staff.

B. Contents of the Submittal

The information listed below should be included in the submittal:

1. For each facility, include the applicable date for the ASME Code and the appropriate addenda date.

2. The period and interval for which this program is applicable.

3. Provide the proposed codes and addenda to be used for repairs, modifications, additions or alternations to the facility which might be implemented during this inspection period.

4. Indicate the components and lines that you have exempted under the rules of Section XI of the ASME Code. A reference to the applicable paragraph of the code that grants the exemption is necessary. The inspection requirements for exempted components should be stated (e.g., visual inspection during a pressure test).

5. Identify the inspection and pressure testing requirements of the applicable portion of Section XI that are deemed impractical because of the limitations of design, geometry, or materials of construction of the components. Provide the information requested in the following section of this appendix for the inspections and pressure tests identified in Item 4 above.

C. Request for Relief from Certain Inspection and Testing Requirements

It has been the staff's experience that many requests for relief from testing requirements submitted by applicants and licensees have not been supported by adequate descriptive and detailed technical information. This detailed information is necessary to: (1) document the impracticality of the ASME Code requirements within the limitations of design, geometry, and materials of construction of components; and (2) determine whether the use of alternatives will provide an acceptable level of quality and safety.

Relief requests submitted with a justification such as "impractical," "inaccessible," or any other categorical basis, require additional information to permit the staff to make an evaluation of that relief request. The objective of the guidance provided in this section is to illustrate the extent of the information that is required by the NRC staff to make a proper evaluation and to adequately document the basis for granting the relief in the staff's Safety Evaluation Report. The NRC staff believes subsequent requests for additional information and delays in completing the review can be considerably reduced if this information is provided initially in the applicant's submittal.

For each relief request submitted, the following information should be included:

1. An identification of the component(s) and/or the examination requirements for which relief is requested.

2. The number of items associated with the requested relief.

3. The ASME Code class.

4. An identification of the specific ASME Code requirement that has been determined to be impractical.

5. The information to support the determination that the requirement is impractical; i.e., state and explain the basis for requesting relief.

6. An identification of the alternative examinations that are proposed: (a) in lieu of the requirements of Section XI; or (b) to supplement examinations performed partially in compliance with the requirements of Section XI.

7. A description and justification of any changes expected in the overall level of plant safety by performing the proposed alternative examinations in lieu of the examination required by Section XI. If it is not possible to perform alternate examinations, discuss the impact on the overall level of plant quality and safety.

For inservice inspection, provide the following additional information regarding the inspection frequency:

8. State when the request for relief would apply during the inspection period or interval (i.e., whether the request is to defer an examination).

9. State when the proposed alternative examinations will be implemented and performed.

10. State the time period for which the requested relief is needed.

Technical justification or data must be sumitted to support the relief request. Opinions without substantiation that a change will not affect the quality level are unsatisfactory. If the relief is requested for inaccessibility, a detailed description or drawing which depicts the inaccessibility must accompany the request. A relief request is not required for tests prescribed in Section XI that do not apply to your facility. A statement of "N/A" (not applicable) or "None" will suffice.

D. Request for Relief for Radiation Considerations

Exposures of test personnel to radiation to accomplish the examinations prescribed in Section XI of the ASME Code can be an important factor in determining whether, or under what conditions, an examination must be performed. A request for relief must be submitted by the licensee in the manner described above for inaccessibility and must be subsequently approved by the NRC staff.

We recognize that some of the radiation considerations will only be known at the time of the test. However, the licensee generally is aware, from experience at operating facilities, of those areas where relief will be necessary and should submit as a minimum, the following information with the request for relief:

1. The total estimated man-rem exposure involved in the examination.

2. The radiation levels at the test area.

3. Flushing or shielding capabilities which might reduce radiation levels.

4. A proposal for alternate inspection techniques.

5. A discussion of the considerations involved in remote inspections.

6. Similar welds in redundant systems or similar welds in the same systems which can be inspected.

7. The results of preservice inspection and any inservice results for the welds for which the relief is being requested.

8. A discussion for the consequences if the weld which was not examined, did fail.

PRESERVICE AND INSERVICE INSPECTIONS-Snubbers

TO ALL APPLICANTS:

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9 This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a mimimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifictions.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1,4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250° F should be verified as follows:

(a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for .he pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

Containment Sump and its effect on long term cooling following a LOCA

During our reviews of license applications we have identified concerns related to the containment sump design and its effect on long term cooling following a Loss of Coolant Accident (LOCA).

These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core, (2) inadequate NPSH of the pumps taking suction from the containment sump, (3) air entrainment from streams of water or steam which can cause loss of adequate NPSH, (4) formation of vortices which can cause loss of adequate NPSH, air entrainment and suction of floating debris into the ECCS and (5) inadequate emergency procedures and operator training to enable a correct response to these problems. Preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications.

The NRC has begun a generic program to resolve this issue. However, more immediate actions are required to assure greater reliability of safety system operation. We therefore require you take the following actions to provide additional assurance that long term cooling of the reactor core can be achieved and maintained following a postulated LOCA.

1. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable

inspection lights, unsecured wood, construction materials and tools as well as other miscellaneous loose equipment. "As licensed" cleanliness should be assured prior to each startup.

This inspection shall be performed at the end of each shutdown as soon as practical bef..e containment isolation.

2. Institute an inspection program according to the requirements of Regulatory Guide 1.82, item 14. This item addresses inspection of the containment sump components including screens and intake structures.

3. Develop and implement procedures for the operator which address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These procedures should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

4. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

   Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

5. Evaluate the extent to which the containment sump(s) in your plant meet the requirements for each of the items previously identified; namely

debris, inadequate NPSH, air entrainment, vortex formation and operator actions.

The following additional guidance is provided for performing this evaluation.

(1) Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.

(2) Provide a drawing showing the location of the drain sump relative to the containment sumps.

(3) Provide the following information with your evaluation of debris:

(a) Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump (or torus), the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

(b) Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).

(c) For each type of thermal insulation used in the containment, provide the following information:

(i) type of material including composition and density,

(ii) manufacturer and brand name,

(iii) method of attachment,

(iv) location and quantity in containment of each type,

(v) an estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

(d) Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS

260.17 Section 17.1.2.2 of the standard format (Regulatory Guide 1.70) requires the identification of safety-related structures, systems, and components (Q-list) controlled by the QA program. You are requested to supplement and clarify the Diablo Canyon Q-list in Table 3.2-4 of the FSAR in accordance with the following:

a. The following items do not appear on the Q-list (FSAR Table 3.2-4). Add the appropriate items to the Q-list and provide a commitment that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

1. Safety-related masonry walls (see IE Bulletin No. 80-11).
2. Breakwaters.
3. Leak detection system (see FSAR Section 3.5).
4. Missile barriers which protect safety-related items.
5. Onsite power system (Class 1E).
   a) Electrical penetrations of containment - Non-vital including primary and backup fault current protective devices.
   b) Raceway fire stops and seals.
   c) Emergency light battery packs.
6. Radiation monitoring (fixed and portable).
7. Radioactivity monitoring (fixed and portable).
8. Radioactivity sampling (air, surfaces, liquids).
9. Radioactive contamination measurement and analysis.
10. Personnel monitoring internal (e.g., whole body counter) and external (e.g., TLD system).
11. Instrument storage, calibration, and maintenance.
12. Decontamination (facilities, personnel, and equipment).
13. Respiratory protection, including testing.
14. Contamination control.
15. Radiation shielding.
16. Meteorological data collection programs.
17. Expendable and consumable items necessary for the functional performance of safety-related structures, systems, and components (i.e., weld rod, fuel oil, boric acid, snubber oil, etc.).

18. Measuring and test equipment used for safety-related structures, systems, and components.

19. Ground slope east of building complex.

20. Firewater storage reservoir ponds.

21. Hydrogen recombiner, including piping and valves.

22. Containment pressure indication system.

23. Containment water level indication systems.

24. Containment hydrogen indication system.

25. Valve operators for safety-related valves.

26. Motors for safety-related pumps.

b. The following items from the Q-list (FSAR Table 3.2-4) need expansion and/or clarification as noted. Revise the list as indicated or justify not doing so.

1. Portions of the turbine generator building (sheet 4) which enclose the emergency diesel-generator units and ancillary systems as well as other safety-related components should be under the controls of the operational QA program.

2. New fuel storage racks (sheet 3) should be under the controls of the operational QA program.

3. Intake structure and conduit (sheet 5) should be under the controls of the operational QA program.

4. Containment structure sump, sump screen, and vortex suppression should be under the controls of the operational QA program.

5. Reactor cavity sump pump (sheet 18) should be under the controls of the operational QA program.

6. Clarify that the primary system PORV, safety valves, and PORV block valves and their actuators are included under "Reactor Coolant Systems Valves," (sheet 25).

7. Clarify that the main steamline safety valves and steamline PORVs and their actuators are included under "Valves for the Above (Main Steam Piping-SG to MSIV) Portion of System" (sheet 23).

8. Identify the safety-related instrumentation and control systems to the same scope and level of detail as provided in Chapter 7 of the FSAR.

9. The 250V DC Motor Control Center SD 121 (sheet 36) should be under the controls of the operational QA program.

10. Circulating water conduits (sheet 5) should be under the controls of the operational QA program.

c. Enclosure 2 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (November 1980) identified numerous items that are safety-related and appropriate for OL application and therefore should be on the Q-list. These items are listed below. Add the appropriate items to the Q-list and provide a commitment that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

	NUREG-0737 (Enclosure 2) Clarification Item
1) Plant-safety-parameter display console.	I.D.2
2) Reactor coolant system vents.	II.B.1
3) Plant shielding.	II.B.2
4) Post accident sampling capabilities.	II.B.3
5) Valve position indication.	II.D.3
6) Auxiliary feedwater system.	II.E.1.1
7) Auxiliary feedwater system initiation and flow.	II.E.1.2
8) Emergency power for pressurizer heaters.	II.E.3.1
9) Dedicated hydrogen penetrations.	II.E.4.1
10) Containment isolation dependability.	II.E.4.2
11) Accident monitoring instrumentation.	II.F.1
12) Instrumentation for detection of inadequate core-cooling.	II.F.2
13) Power supplies for pressurizer relief valves, block valves, and level indicators.	II.G.1
14) Automatic PORV isolation.	II.K.3(1)
15) Automatic trip of reactor coolant pumps.	II.K.3(5)
16) PID controller.	II.K.3(9)
17) Anticipatory reactor trip on turbine trip.	II.K.3(12)
18) Power on pump seals.	II.K.3(25)
19) Emergency plans.	III.A.1.1/III.A.2
20) Emergency support facilities.	III.A.1.2
21) Inplant I_2 radiation monitoring.	III.D.3.3
22) Control-room habitability.	III.D.3.4

420.0 INSTRUMENTATION AND CONTROL SYSTEMS

420.5 (7.0) (15.0) Loss of Non-Class 1E Instrumentation and Control Power System Bus During Power Operation (IE Bulletin No. 79-27)

If reactor controls and vital instruments derive power from common electrical distribution systems, the failure of such electrical distribution systems may result in an event requiring operator action concurrent with failure of important instrumentation upon which these operator actions should be based. This concern was addressed in IE Bulletin No. 79-27. On November 30, 1979, IE Bulletin No. 79-27 was sent to operating license (OL) holders, the near term OL applicants (North Anna 2, Diablo Canyon, McGuire, Salem 2, Sequoyah, and Zimmer), and other holders of construction permits (CP), including the Shearon Harris Nuclear Power Plant, Units 1, 2, 3 & 4. Of these recipients, the CP holders were not given explicit direction for making a submittal as part of the licensing review. However, they were informed that the issue would be addressed later.

You are requested to address this issue by taking IE Bulletin No. 79-27 Actions 1 thru 3 under "Actions to be Taken by Licensees". Within the response time called for in the attached transmittal letter, complete the review and evaluation required by Actions 1 thru 3 and provide a written response describing your reviews and actions. This report should be in the form of an amendment to your FSAR and submitted to the NRC Office of Nuclear Reactor Regulation as a licensing submittal.

420.6 (7.3) Engineered Safety Features (ESF) Reset Controls (IE Bulletin No. 80-06)

If safety equipment does not remain in its emergency mode upon reset of an engineered safeguards actuation signal, system modification, design change or other corrective action should be planned to assure that protective action of the affected equipment is not compromised once the associated actuation signal is reset. This issue was addressed in IE Bulletin No. 80-06. For facilities with operating licenses as of March 13, 1980, IE Bulletin No. 80-06 required that reviews be conducted by the licensees to determine which, if any, safety functions might be unavailable after reset, and what changes could be implemented to correct the problem.

For facilities with a construction permit including OL applicants, IE Bulletin No. 80-06 was issued for information only.

The NRC staff has determined that all CP holders, as a part of the OL review process, are to be requested to address this issue. Accordingly, you are requested to take the actions called for in IE Bulletin No. 80-06 Actions 1 thru 4 under "Actions to be Taken by Licensees". Within the response time called for in the attached transmittal letter, complete the review verifications and descriptions

420.6 (7.3) (Cont.)

of corrective actions taken or planned as stated in Actions 1 thru 3 and submit the report called for in Action Item 4. The report should be submitted to the NRC Office of Nuclear Reactor Regulation as a licensing submittal in the form of an FSAR amendment.

420.7 (7.0) (15.0)

Qualification of Control Systems (IE Information Notice No. 79-22)

Operating reactor licensees were informed by IE Information Notice No. 79-22, issued September 19, 1979, that certain non-safety grade or control equipment, if subjected to the adverse environment of a high energy line break, could impact the safety analyses and the adequacy of the protection functions performed by the safety grade equipment. Attached is a copy of IE Information Notice No. 79-22, and reprinted copies of an August 20, 1979 Westinghouse letter and a September 10, 1979 Public Service Electric and Gas Company letter which address this matter. Operating Reactor licensees conducted reviews to determine whether such problems could exist at operating facilities.

We are concerned that a similar potential may exist at light water facilities now under construction. You are, therefore, requested to perform a review to determine what, if any, design changes or operator actions would be necessary to assure that high energy line breaks will not cause control system failures to complicate the event beyond your FSAR analysis. Provide the results of your reviews including all identified problems and the manner in which you have resolved them to NRR.

The specific "scenarios" discussed in the above referenced Westinghouse letter are to be considered as examples of the kinds of interactions which might occur. Your review should include those scenarios, where applicable, but should not necessarily be limited to them.

420.8 (7.0) (15.0)

Control System Failures

The analyses reported in Chapter 15 of the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents.

Based on the conservative assumptions made in defining these design-basis events and the detailed review of the analyses by the staff, it is likely that they adequately bound the consequences of single control system failures.

To provide assurance that the design basis event analyses adequately bound other more fundamental credible failures, you are requested to provide the following information:

420.8
(7.0)
(15.0)
(Cont.)

(1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

(3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors. The sensors considered should include, but should not necessarily be limited to, common hydraulic headers or impulse lines feeding pressure, temeprature, level or other signals to two or more control systems.

(4) Provide justification that any simultaneous malfunctions of the control systems identified in (2) and (3), resulting from failures or malfunctions of the applicable common power source or sensor, are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

UN ED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

September 14, 1979

IE Information Notice No. 79-22

QUALIFICATION OF CONTROL SYSTEMS

Public Service Electric and Gas Company notified the NRC of a potential unreviewe safety question at their Salem Unit 1 facility. This notification was based on a continuing review by Westinghouse of the environmental qualifications of equipmen that they supply for nuclear steam supply systems. Based on the present status of this effort, Westinghouse has informed their customers that the performance of non-safety grade equipment subjected to an adverse environment could impact the protective functions performed by safety grade equipment. These non-safety grade systems include:

Steam generator power operated relief valve control system

Pressurizer power operated relief valve control system

Main feedwater control system

Automatic rod control system

These systems could potentially malfunction due to a high energy line break inside or outside of containment. NRC is also concerned that the adverse environment could also give erroneous information to the plant operators. Westinghouse states that the consequences of such an event could possibly be more limiting than results presented in Safety Analysis Reports, however, Westinghouse also states that the severity of the results can be limited by operator actions together with operating characterisitics of the safety systems. Further, Westinghouse has recommended to their customers that they review their systems to determine whether any unreviewed safety questions exist.

This Information Notice is provided as an early notification of a possibly significant matter. It is expected that recipients will review the information for possible applicability to their facilities. No specific action or response is requested at this time. If NRC evaluations so indicate, further licensee actions may be requested or required. If you have questions regarding this matt please contact the Director of the appropriate NRC Regional Office.

No written response to this Information Notice is required.

REPRINT

Westinghouse Electric Corporation
Water Reactor Division
Nuclear Service Division
Box 2728
Pittsburgh, Pennsylvania 15230

August 30, 1979
PSE-79-21

Mr. F. P. Librizzi, General Manager
Electric Production
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

Dear Mr. Librizzi:

Public Service Electric and Gas Co.
Salem Unit No. 1
QUALIFICATION OF CONTROL SYSTEMS

As part of a continuing review of the environmental qualifications of Westinghouse supplied NSSS equipment, Westinghouse has also found it necessary to consider the interaction with non-safety grade systems. This investigation has been conducted to determine if the performance of non-safety grade systems which may not be protected from an adverse environment could impact the protective functions performed by NSSS safety grade equipment. The NSSS control and protection systems were included in this review to assess the adequacy of the present environmental qualification requirements.

As a result of this review, several systems were identified which, if subjected to an adverse environment, could potentially lead to control system operation which may impact protective functions. These systems are:

- Steam generator power operated relief valve control system
- Pressurizer power operated relief valve control system
- Main feedwater control system
- Automatic rod control system

Each of the above mentioned systems could potentially malfunction if impacted by adverse environments due to a high energy line break inside or outside containment. In each case, a limited set of breaks, coupled with possible consequential control malfunction in an adverse direction, of the above events could yield results which are more limiting than those presented in the plant Safety Analysis Reports. In all cases, however, the severity of the results can be limited by operator actions together with operating characteristics of the safety systems.

We believe these systems identified do not constitute a substantial safety hazard. However, Westinghouse recommends you review them to determine if any unreviewed safety questions or significant deficiencies exist in your plant(s).

To assist you in understanding these concerns, Westinghouse will hold a seminar in Pittsburgh on Thursday, September 6 at Westinghouse R&D Center, Building 701, with all our operating plant customers. The seminar will address the potential impact of these concerns for various plant designs and various licensing bases.

Please contact your WNSD Regional Service office to confirm your attendance at the seminar. We will provide additional details concerning the agenda and other meeting arrangements as they become available.

Very truly yours,

ORIGINAL SIGNED BY

F. Noon, Manager
Eastern Regional & WNI Support

SR4/CC13&14

cc: H. J. Midura
H. J. Heller
R. D. Rippe
T. N. Taylor
R. A. Uderitz
C. F. Barclay W

ATTACHMENT 420-3

REPRINT

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
Salem Nuclear Generating Station
P. O. Box 56
Hancocks Bridge, New Jersey 08038

September 10, 1979

Mr. Boyce H. Grier
Director of USNRC
Office of Inspection and Enforcement
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Sir:

REPORTABLE OCCURRENCE 79-58/01P
SALEM NO. 1 UNIT LER

This letter will serve to confirm our telephone report to Mr. Gary Schneider of the Regional NRC office on Friday, September 6, 1979, advising of a potential reportable occurrence in accordance with Technical Specification 6.9.1.8.

We have been notified by our Engineering Department that a Westinghouse conducted review of the environmental qualifications of Westinghouse supplied NSSS equipment has identified that conditions associated with high energy line breaks inside or outside containment and their impact on non-safety control systems may constitute an unreviewed safety question. The control systems concerned are steam generator power operated relief valve control, pressurizer power operated relief valve control, main feedwater control and automatic rod control systems.

A detailed report will be submitted in the time period specified by the Technical Specifications.

Very truly yours,

Original Signed By

H. J. Midura
Manager - Salem Generating Station

AWK:jds

CC: General Manager - Electric Production
Manager - Quality Assurance