GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

June 30, 1993

MFN No. 103-93
Docket No. STN 52-004

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Richard Borchardt, Acting Director
Standardization Project Directorate

Subject: **NRC Request for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design**

Reference:
1. Transmittal of Requests for Additional Information (RAIs) for the SBWR Design, Letter from J. W. Thompson to P. W. Marriott, January 28, 1993
2. NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design, Letter from P. W. Marriott to Richard Borchardt, MFN No. 052-93, April 9, 1993
3. NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design, Letter from P. W. Marriott to Richard Borchardt, MFN No. 053-93, April 9, 1993
4. NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design, Letter from P. W. Marriott to Richard Borchardt, MFN No. 083-93, May 28, 1993

The reference requested additional information on the SBWR Design. As part of the response to this request, GE submitted responses in References 2 - 4.

This letter transmits the remainder of the responses to these RAIs. In addition, to provide more complete information, such as SSAR changes, this letter provides revised responses to the following RAIs transmitted by Reference 4: ECGB.21, EMCB.4, .7 and .10, and SRXB.28 and .45.

Sincerely,

R.C. Mitchell
for D. J. Robare, Acting Manager
Safety & Licensing
M/C 481, (408) 925-3141

Enclosure: RAI Responses

LTRBK 93-36

010045

D040

9307020176 930630
PDR ADOCK 05200004
A PDR

bcc:	J. C. Baechler	(GE)
	J. A. Beard	(GE)
	R. W. Burke, Sr.	(EPRI)
	S. A. Delvin	(GE)
	D. L. Foreman	(GE)
	D. M. Gluntz	(GE)
	P. W. Marriott	(GE)
	A. S. Rao	(GE)
	F. A. Ross	(DOE)
	SBWR File MC-781	

RAI Number: ECGB.1

Question:

Provide the justification as to why no information was provided for the design features described in SSAR Sections 2.5.5, "Stability of Slopes," and 2.5.6, "Embankment and Dams."

GE Response:

The combined operating license (COL) applicant will provide the site characteristics information in accordance with 10CFR52.79. Sections 2.5.5, "Stability of Slopes," and 2.5.6, "Embarkment and Dams," are site characteristics information and will be revised in Amendment 1 of the SSAR to indicate that the information will be provided by the COL applicant (see attached).

2.5.5 Stability of Slopes

~~None.~~

Stability of slopes is a site characteristic information, and shall be provided by the COL applicant. (See Subsection 2.7.1.)

2.5.6 Embankments and Dams

~~None.~~

Embankments and dams are site characteristic information, and shall be provided by the COL applicant. (See Subsection 2.7.1.)

2.6 Requirements for Determination of Site Acceptability

This section provides the requirements for the determination of SBWR site acceptability. Acceptability is required from the standpoint of both design bases events and severe accidents.

2.6.1 Design Basis Events

For design bases events, the site is acceptable if all of the site characteristics fall within the envelope of the SBWR Standard Plant site design parameters provided in Sections 2.3, 2.4, and 2.5. For cases where a characteristic exceeds its envelope, it will be necessary for the COL applicant to submit analyses to demonstrate that the overall set of site characteristics do not exceed the capability of the design. (See Subsection 2.7.6 for COL applicant license information requirements.)

2.6.2 Severe Accidents

The SBWR probabilistic risk assessment (PRA) results are calculated for an average or typical site, as outlined in Appendix 19E. Although these results form a good basis for assessing the general SBWR capability to satisfy off-site dose-related goals, they do not form a basis for concluding that the SBWR would meet dose related goals at a specific site whose characteristics cannot be defined at the point of SBWR certification. Consistent with the certification concept that all key technical issues be resolved before certification, it is appropriate to define the process for determining future site acceptability. This process is defined below in terms of (1) acceptance, (2) data input, and (3) analysis.

Acceptance Criteria

Site acceptability for severe accidents will be based upon a calculation using the MACCS computer code for determination of SBWR site acceptability. The results of such a calculation will be compared to the goals of Appendix 19E as shown in Table 2.6-1. The site will be deemed acceptable if the results fall within the given goals.

RAI Number: ECGB.2

Question:

In SSAR Table 3.2-1, Item B21.6, the feedwater line classification break from Quality Group (QG) B to QG D is located at the seismic interface restraint. In Figure 21.5.1-1, this break is located at the shut-off valve. The staff's position is that this classification break should be at the seismic restraint. Revise Figure 21.5.1-1 to agree with Table 3.2-1.

GE Response:

Although the seismic restraint locates the interface of the seismic classification, the quality group break at the shut-off valve is consistant with what has previously been agreed upon in the ABWR design. Table 3.2-1, Item B21.8, specifically addresses this section of FW piping. Thus, the Table and the Figure are in agreement.

RAI Number: ECGB.3

Question:

SSAR Section 6.7 states that the SBWR alternate to a main steam isolation valve leakage control system is contained in Appendix 19H. Since this Appendix will not be submitted until February 28, 1993, the staff cannot complete its review of this issue. However, Table 3.2-1 appears to contain acceptable commitments to the staff positions relative to the structural integrity of piping systems and components applicable to this issue with the exception of the following:

a) If the proposed alternate leakage path contains both the main steam drain lines and the turbine by-pass lines, Items B21.13 and N37 in Table 3.2-1 should contain a commitment that these lines will be dynamically analyzed for the safe shutdown earthquake (SSE) up to the condenser. This same commitment should be added to Section 10.4.4 for the turbine by-pass lines. Enclosure

b) Table 3.2-1, Item N61 and Subsection 10.4.1 should both contain a commitment that the condenser anchorage is dynamically analyzed for the SSE.

GE Response:

The alternate leakage path for the main steam isolation valve contains both the main steam drain lines and the turbine bypass lines, Items B21.13 and N37 in Table 3.2-1. These lines will be analysed for SSE seismic loading, and appropriate sections revised in Amendment 1 of the SSAR.

The condenser anchorage will be analyzed for SSE seismic loading, and Table 3.2-1, Item N61, and Subsection 10.4.1 will be revised in Amendment 1 of the SSAR to reflect this commitment.

RAI Number: ECGB.4

Question:

In SSAR Table 3.2-1, Item C61, Remote Shutdown System (RSS), is classified as not safety-related and non-seismic. It is stated in this table that the RSS controls some components that are in the control rod drive (CRD), reactor water cleanup (RWCU)/shutdown cooling (SDC), reactor component cooling water system (RCCWS), and heating, ventilation, and air conditioning (HVAC). Subsection 7.4.2 in the SSAR states that the RSS does not include control interfaces with safety-related equipment. In the advanced boiling water reactor (ABWR), the RSS is Safety Class 3, quality assurance (QA) B, and seismic Category I. Since some of the components controlled by the RSS in the SBWR may be safety-related, provide the basis for the non-safety and non-seismic classifications for the RSS.

GE Response:

In SSAR Table 3.2-1, Item C61, Remote Shutdown System (RSS), is classified as not safety-related and non-seismic because (1) the RSS does not perform any safety-related functions and (2) a failure in the RSS will not prevent any safety-related functions. Shutdown from outside the control room is not a design basis event (DBE). Therefore, per the regulatory definition of safety-related structures, systems, and components in 10CFR50.49(b)(1) and 10 CFR 100, Appendix A, VI(a)(1), the systems and components that are relied upon for the remote shutdown outside the control room do not perform a safety-related function and need not be classified safety-related or seismic Category I. The portions of other systems (the Control Rod Drive [CRD]; Reactor Water Cleanup [RWCU]/Shutdown Cooling [SDC]; Reactor Component Cooling Water System [RCCWS]; and heating, ventilation, and air conditioning [HVAC]) that the RSS controls and interfaces with are not safety-related. Some portions of these systems are safety-related (like the RWCU/SDC); however, the valves in the RWCU/SDC System have local control where each valve has a local panel control switch, from the safety division system switch gear to which they belong. The controls are considered a part of the RSS but are not located in the RSS main panel. In theABWR, the RSS controls and interfaces with safety-related portions of other systems; therefore it is classified safety-related and seismic Category I.

RAI Number: ECGB.5

Question:

In SSAR Table 3.2-1, Items E50.2 and E50 the piping and valves (including supports) in the gravity driven cooling system (GDCS), from the check valves upstream of the squib valves to the suppression and GDCS pools and from the GDCS pools to the lower drywell are QG C. According to Section 6.3 in the SSAR, the GDCS is considered to be one of the SBWR emergency core cooling systems. Therefore, in accordance with Regulatory Position C.1.a of Regulatory Guide (RG) 1.26, this portion of the GDCS should be classified as QG B. Either revise Items E50.2 and E50.3 in Table 3.2.1 and applicable portions of Figure 21.5.3-2 in the SSAR to agree with the staff position, or provide the basis for the QG C classification.

GE Response:

In SSAR Table 3.2-1, Items E50.2 and E50.3, the piping and valves (including supports) in the gravity driven cooling system (GDCS), from the check valves upstream of the squib valves to the suppression and GDCS pools and from the GDCS pools to the lower drywell, are classified as Quality Group C. This classification is in accordance with the definitions of Quality Groups discussed in SSAR Section 3.2.2. The GDCS is an emergency core cooling system as discussed in SSAR Section 6.3.

Although RG 1.26 is used as a guide in defining quality groups for the SBWR, it is not directly applicable to the SBWR design in all cases. The definitions of the Quality Groups provided in RG 1.26 are based on, and specifically address, the LWR designs that were developed in the late 1960s and early 1970s. The SBWR design is a major departure from those earlier designs. Of specific concern is the application of the RG 1.26 guidelines to the GDCS component classification. When RG 1.26 was written, emergency core cooling systems in LWRs extended beyond the primary containment boundary, and, therefore, during system operation, contained primary system water without the protective envelope of the containment structure. These cooling systems were classified as Quality Group B. Conversely, RG 1.26 classified cooling water systems supporting the emergency core cooling systems as Quality Group C, even though these supporting systems had the same functions and importance as the emergency core cooling systems. The supporting systems were only fundamentally different from the emergency core cooling systems in that they did not process potentially contaminated primary system water outside of the containment. Therefore, it is concluded that the higher classification of traditional emergency core cooling systems was not due to their system function, but due to their configuration extending beyond the containment boundary and processing primary system water.

Based on the above discussion, the GDCS components in question are classified as Quality Group C, rather than Quality Group B, because the system configuration does not extend beyond the containment boundary.

It should be noted that classification of GDCS components in the SSAR is consistent with the definitions provided in Draft 9 of the proposed American National Standard ANS-58.14, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors."

RAI Number: ECGB.6

Question:

In SSAR Table 3.2-1, Item G21.4 of the fuel and auxiliary pools cooling system, all of the piping and valves between inboard containment isolation valves and their termination points inside containment are classified as non-safety, QG D, no quality assurance requirement, and seismic Category II. Some of these classifications are not totally consistent with applicable portions of the ABWR. However, the discussions in Subsections 6.2.1.1, "Pressure Suppression Containment," and 6.2.2, "Passive Containment Cooling System," (PCCS) imply that the PCCS performs the safety-related functions of some of those systems listed in Item G21.4. Provide a more detailed discussion of the bases for the classifications in Item G21.4.

GE Response:

The piping, valves and supports specified in the SSAR Table 3.2-1, Item G21.4, from their inboard containment isolation valves to their termination points inside the containment are classified as non-safety, QC D (no quality assurance requirement), and seismic Category II. The referenced piping is part of the Fuel and Auxiliary Pools Cooling System (FAPCS-G21 System) as follows:

- GDCS pools suction line
- GDCS pools return line
- wetwell spray line
- drywell spray line
- suppression pool return line
- reactor well and head cavity drain line

The differences in classification for similar piping systems in the ABWR and SBWR result from the use of different systems to handle the consequences of accidents and hazards.

In order to further discuss the perceived discrepancies of safety functions between those systems listed in Item G21.4 and Subsections 6.2.1.1, "Pressure Suppression Containment" and 6.2.2, "Passive Containment Cooling System" it is recommended that the reviewer refer to the following Figures:

Fig. 21.6.2-1 "Passive Containment Cooling System P&ID."
Fig. 21.9.1-1 Sheet 1 "Fuel & Auxiliary Pools Cooling System P&ID."
Fig. 21.9.1-1 Sheet 2 "Fuel & Auxiliary Pools Cooling System P&ID."
Fig. 21.9.1-1 Sheet 3 "Fuel & Auxiliary Pools Cooling System P&ID."

The Gravity Drain Cooling System (GDCS) pools suction and return lines, and the suppression pool return line are part of the FAPCS which cools and cleans the water in the GDCS and suppression pools and also supplies makeup water during normal operations. The GDCS and suppression pools are designed to operate for all design basis events without requiring water makeup.

The wetwell and drywell spray lines are not required to operate during the postulated design basis events as the SBWR does not take credit for these systems to handle the consequences of design basis accident scenarios.

The portion of the reactor well and head cavity line between the isolation valve and the suppression pool is not required to mitigate any design basis scenario, and if failure is postulated during refueling the isolation valve will prevent drainage of the reactor head cavity volume.

The portion of the FAPCS described in Section 6.2.2.2.2 that provides a dedicated safety related makeup water supply to the Isolation Condenser/Passive Containment Cooling Condenser pool and is shown on Figure 21.9.1-1 Sheet 3.

RAI Number: ECGB.7

Question:

In SSAR Table 3.2-1, Item G21.6, piping and valves between the low-pressure coolant injection (LPCI) gate valve (F332 on Figure 21.9.1-1, Sh. 2) and the interface with the RWCU/shutdown cooling system (SCS) is shown as non-safety-related and no QG classification. On Figure 21.9.1-1, this portion of piping is shown as safety-related, QG B (8"-FD-B) and it connects to an 8"-FD-B line in the RWCU/SDC system (Ref. Figure 21.5.4-2, Sh. 2). Revise Item G21.6 to agree with the classifications in Figure 21.9.1-1.

GE Response:

In the SSAR Table 3.2-1, Item G 21.6, the classification of piping and valves between the low-pressure coolant injection (LPCI) gate valve (F332 on Figure 21.9.1-1 Sheet 2) and the interface with the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SCS) will be revised in Amendment 1 (see attached). The revised classification of this section of piping and valves will be Safety Designation Q and Quality Group B.

Table 3.2-1 Classification Summary (Continued)

Principal Component[1]	Safety Desig.[2]	Location[3]	Quality Group[4]	QA Req.[5]	Seismic Category[6]	Notes
4. Piping and valves including supports between inboard containment isolation valves and their termination points inside containment, for – GDCS pools suction line – GDCS pools return line – wetwell spray line – drywell spray line – suppression pool return line – reactor well & head cavity drain line	N	CV	D	—	II	
5. Interconnecting piping between GDCS pools	N	CV	D	—	II	
6. Piping and valves including supports between low pressure coolant injection gate valve (including valve) up to the interface with Reactor Water Cleanup/Shutdown Cooling System	~~N~~Q	RB	~~—~~B	B	I	
7. All other mechanical modules and piping, including normal makeup system components	N	SE,OO, RB	D	E	NS	
8. Electrical modules and cables with safety-related function	Q	RB,CE, CV	—	B	I	
9. Electrical modules and cables with non-safety-related function	N	RB,CE	—	E	NS	
G31 Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC)						
1. Piping including supports and valves within and including outermost containment isolation valves on pump suction	Q	CV, SE	A	B	I	(7)

RAI Number: ECGB.8

Question:

In SSAR Table 3.2-1, Items K and U74, Radioactive Waste Management Systems and Radwaste Building Structure, commitments are made that a quality assurance program meeting the guidance of RG 1.143 is applied to all of the non-safety items in these systems and structures. In addition, commitments to Section 5, "Seismic Design for Radwaste Management Systems and Structures Housing Radwaste Management Systems," in RG 1.143 should be made in this table for both Items K and U74. Since the SBWR does not include the OBE as a design requirement, provide the seismic design criteria that will be implemented to conform to Section 5 of RG 1.143.

GE Response:

Plant structures, systems, components, and parts shall be designed to the seismic requirements of the Unified Building Code (UBC), Zone 2A, with the exception of those classified as Seismic I or II or those requiring a higher level of seismic design for investment protection and defense-in-depth as defined in Paragraph 2.18. Building structures shall be classified per the UBC as "essential facilities," i.e., with an importance factor of 1.25 for seismic design. Either of the methods permitted by UBC, simplified analysis or dynamic analysis, is acceptable for determination of seismic loads on non-service structures and equipment.

RAI Number: ECGB.9

Question:

In SSAR Table 3.2-1, Item U73, the stack is classified as non-safety and non-seismic. In Figure 21.1.2-2, Sh. 2, the stack appears to be a part of the reactor building outer shell. In Section 3.8.4.1, the reactor building outer shell is identified as seismic Category I. Either revise Item U73 to classify the stack as safety-related and seismic Category I, or provide the basis for the non-safety and non-seismic classifications.

GE Response:

The stack is classified as non-seismic. It shall be designed and constructed according to the seismic requirements of the Unified Building Code (UBC), Zone 2A. It shall be classified per the UBC as "essential facilities," i.e., with an importance factor of 1.25 for seismic design. Either of the methods permitted by UBC, simplified analysis or dynamic analysis, is acceptable for determination of seismic loads on non-service structures and equipment. However, the stack has been included in the Reactor Building seismic analysis model for determination of seismic loads. The stack shall be anchored adequately into the Reactor Building structure to preclude its collapse onto a Seismic Category I structure.

RAI Number: ECGB.10

Question:

In SSAR Subsection 3.6.2.1.1, in the paragraph on page 3.6-13 entitled "Non-ASME Class Piping," add the following commitment to be consistent with Standard Review Plan (SRP) 3.6.2: "Separation and interaction requirements between seismically analyzed and non-seismically analyzed piping are met as described in Subsection 3.7.3.8."

GE Response:

This comment is covered under 3.6.1.1 page 3.6-3 (5th paragraph). Seismic interaction is also covered in 3.7.3.8.

The recommended sentence will be added to Amendment 1, Subsection 3.6.2.1.1 in the paragraph on page 3.6-13 entitled "Non-ASME Class Piping" (see attached).

ASME Code Section III Class 2 and 3 Piping in Areas Other Than Containment Penetration

With the exceptions of those portions of piping identified above, breaks in ASME Codes, Section III, Class 2 and 3 piping are postulated at the following locations in those portions of each piping and branch run:

- at terminal ends; and
- at intermediate locations selected by one of the following criteria:
  - At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.
  - At each location where stresses calculated by the sum of Equations 9 and 10 in NC/ND-3653, ASME Code, Section III, exceed 0.8 times the sum of the stress limits given in NC/ND3653.

As a result of piping reanalysis caused by differences between the design configuration and the as-built configuration, the highest stress locations may be shifted; however, the initially determined intermediate break locations may be used unless a redesign of the piping resulting in a change in the pipe parameters (diameter, wall thickness, routing) is required, or the dynamic effects from the new (as-built) intermediate break location are not mitigated by the original pipe whip restraints and jet shields.

Non-ASME Class Piping

Breaks in seismically analyzed non-ASME Class (not ASME Class 1, 2, or 3) piping are postulated according to the same requirements for ASME Class 2 and 3 piping above. Separation and interaction requirements between seismically analyzed and non-seismically analyzed piping are met as described in Subsection 3.7.3.8.

Separating Structure With High-Energy Lines

If a structure separates a high energy line from a safety-related component, the separating structure is designed to withstand the consequences of the pipe break in the high-energy line at locations that the aforementioned criteria require to be postulated. However, as noted in Subsection 3.6.1.3, some structures that are identified as necessary by the HELSA evaluation (i.e., based on no specific break locations), are designed for worst-case loads.

3.6.2.1.2 Locations of Postulated Pipe Cracks

Postulated pipe crack locations are selected as follows:

RAI Number: ECGB.11

Question:

In SSAR Subsection 3.6.2.4, it is stated that the SBWR does not require guard pipes. Subsection 3.6.2.1.1 provides criteria for "sleeve assemblies" in the containment penetration areas, and Subsections 5.4.6.3 and 6.2.4.3.2, mention the use of guard pipes in the containment penetrations for the steam supply and condensate lines of the isolation condenser system. In Section 5.4.6.3, it is stated that the design intent for these guard pipes is either to show that the stresses and fatigue usage factors do not exceed special limits in SRP 3.6.2, or to show by proof testing that the guard pipes and transition fittings do not experience crack initiation or crack growth.

a) Revise Subsections 3.6.2.1.1 and 3.6.2.4 to clarify that the criteria for sleeves in Subsection 3.6.2.1.1 are applicable to guard pipes in all containment penetration areas.

b) It is the staff's position that the experimental analysis proof testing briefly discussed in Subsection 5.4.6.3 cannot be used in lieu of the criteria in SRP 3.6.2.

GE Response:

The SBWR does not require guard pipes; therefore, all references to guard pipes have been deleted. Penetration sleeves are used where the isolation condenser supply and return pipes enter the pool at the containment pressure boundary. All SSAR revisions discussed below are shown on the applicable SSAR pages.

a) Subsection 3.6.2.1.1 has been revised to clarify that the criteria for sleeves in Subsection 3.6.2.1.1 are applicable to penetration sleeves in all containment penetration areas.

   In Subsections 5.4.6.2.2, 5.4.6.3, and 6.2.4.3.2, the words "guard pipes" have been replaced with the words "penetration sleeves." Subsection 3.6.2.4 has not been revised because the SBWR does not require guard pipes. Subsection 6.2.4.3.2 has been revised to state that penetration sleeves will meet the requirements of Subsection 3.6.2.1.1.

b) The experimental proof testing briefly discussed in Subsection 5.4.6.3 has been deleted. This section has been revised to state that penetration sleeves are designed and constructed in accordance with the requirements specified in Subsection 3.6.2.1.1.

and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

3.6.2.1.1 Locations of Postulated Pipe Breaks

Postulated pipe locations are selected as follows:

Piping Meeting Separation Requirements
Based on the HELSA evaluation described in Subsection 3.6.1.3, the high-energy lines which meet the spatial separation requirements are generally not identified with particular break points. Breaks are postulated at all possible points in such high-energy piping systems. However, in some systems break points are particularly specified according to the following subsections if special protection devices such as barriers or restraints are provided.

Piping in Containment Penetration Areas
No pipe breaks or cracks are postulated in those portions of piping from the containment ~~wall~~ penetration to and including the inboard or outboard isolation valves which meet the following requirement in addition to the requirement of the ASME Code, Section III, Subarticle NE-1120:

- The following design stress and fatigue limits are not exceeded:

  For ASME Code, Section III, Class 1 Piping

  - The maximum stress range between any two load sets (including the zero load set) does not exceed 2.4 S_m, and is calculated by Equation 10 in NB-3653, ASME Code, Section III.
  - The cumulative usage factor is less than 0.1.
  - The maximum stress as calculated by Equation 9 in NB-3652 under the loadings resulting from a postulated piping failure beyond those portions of piping does not exceed the lesser of $2.25S_m$ and $1.8S_y$ except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stress provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirement identified in Section 3.9.3. Primary loads include those which are deflection limited by whip restraints.

  For ASME Code, Section III, Class 2 Piping

  - The maximum stress as calculated by the sum of Equations 9 and 10 in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits are specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) including an OBE event does not exceed $0.8(1.8\ S_h + S_A)$. The S_h and S_A are allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

– The maximum stress, as calculated by Equation 9 in NC-3653 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping, does not exceed the lesser of $2.25\ S_h$ and $1.8\ S_y$.

Primary loads include those which are deflection limited by whip restraints. The exceptions permitted above may also be applied provided that when the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ANSI B31.1, the piping is either of seamless construction with full radiography of all circumferential welds, or all longitudinal and circumferential welds are fully radiographed.

- Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the above mentioned code limits.

- The number of circumferential and longitudinal piping welds and branch connections are minimized. Where penetration sleeves are used, the enclosed portion of fluid system piping is seamless construction and without circumferential welds unless specific access provisions are made to permit inservice volumetric examination of longitudinal and circumferential welds.

- The length of these portions of piping are reduced to the minimum length practical.

- The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) do not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings may be used) except where such welds are 100% volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the above mentioned code limits.

- Sleeves provided for those portions of piping in the containment penetration areas are constructed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III, where the sleeve is part of the containment boundary. In addition, the entire sleeve assembly is designed to meet the following requirements and tests:

P&ID defines piping system interconnections, valves, instruments, special arrangement requirements, manually operated controls and system input sources and outputs.

Each IC is designed for 30 MWt capacity and is made of two identical modules.

The units are located in a large water pool positioned above and outside the SBWR containment (drywell).

The IC is configured as follows:

- The steam supply line (insulated and enclosed in a ~~guard pipe~~ penetration sleeve which penetrates the containment roof slab) is vertical and feeds two horizontal headers through four pipes. Each pipe is provided with a built-in flow limiter, sized to allow natural circulation operation of the IC at its maximum heat transfer capacity while addressing the concern of IC breaks downstream of the steam supply pipe. Steam is condensed inside vertical tubes and is collected in two lower headers. Two pipes, one from each lower header, take the condensate to the common drain line which vertically penetrates the containment roof slab.
- Vent lines are provided for each upper and lower headers to remove the noncondensable gases away from the unit, during the IC operation period; the lines penetrate the containment roof slab.
- A purge line is provided to assure that, during normal plant operation (IC system standby conditions), the excess of hydrogen (from the hydrogen water chemistry control additions) or air from the feedwater will not accumulate in the IC steam supply line, thus assuring that the IC tubes will not be blanketed with noncondensables when the system is first started. The purge line penetrates the containment roof slab.
- Isolation containment valves are provided on the steam supply piping and the condensate return piping. The containment isolation is discussed in Subsection 6.2.4.
- Located on the condensate return piping just upstream of the reactor entry point is a loop seal and a pair of valves: (1) a condensate return valve (F005, motor-operated, fail as is) and (2) a condensate return bypass valve (F006, nitrogen piston operated, fail open). These two valves are closed during normal station power operations. Since the steam supply line valves are normally open, condensate will form in the IC and will develop a level up to steam distributor, above the upper headers. To start an IC into operation, the motor-operated condensate return valve (F005) is opened, whereupon the standing condensate drains into the reactor and the steam- water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The fail-open nitrogen

If, during IC operation and after the initial transient, the RPV gauge pressure increases above 7.653 MPa (1110 psig), the bottom vent valves (F009 and F010) automatically open; when the RPV gauge pressure decreases below 7.584 MPa (1100 psi) (reset value) and after a time delay to avoid too many cycles, these two valves close.

After reactor isolation and automatic IC System operation, the control room operator can control the venting of noncondensable gases from the IC to enable it to hold reactor pressure below safe shutdown limits.

5.4.6.3 Safety Evaluation

The Isolation Condenser System is used to transfer decay and residual heat from the reactor after it is shutdown and isolated. This function can also be performed by the RWCU/SDC System or Engineered Safety Features (ESF)of ADS, PCCS, and GDCS which back up the ICS. The Isolation Condenser System is designed and qualified as a safety-related system to comply with 10CFR 50 Appendix A, Criterion 34 and to avoid unnecessary use of these ESFs for residual heat removal, but it is not an Engineered Safety Feature.

The ICS parts (including isolation valves) which are located inside the containment and out to the IC flow restrictors are designed to ASME Code Section III, Class 1, Regulatory Guide 1.26, Quality Group A. The ICS parts which are located outside the containment downstream of the flow restrictor are designed to ASME Code Section III, Class 2, Regulatory Guide 1.26, Quality Group B. The electrical design systems are designed to comply with Class 1E requirements per Regulatory Guide 1.153, and the entire system is designed to Seismic Category I per Regulatory Guide 1.29.

The common IC/PCC pool that ICs share with the PCCs of the Passive Containment Cooling System is safety-related and an ESF because of the PCC function (Subsection 6.2.2.1).

Two out of three ICS loops will remove post-reactor isolation decay heat and depressurize the reactor to safe shutdown conditions when the reactor is isolated after operation at 100% power and with loss of feedwater makeup to the reactor. One ICS loop will be capable of removing decay heat and depressurize the reactor when isolated with continued feedwater or CRD makeup.

As protection from missile, tornado and wind, the ICS parts outside the containment (the Isolation Condenser itself) are located in a subcompartment of the safety-related ICS/PCC pool to comply with 10CFR 50 Appendix A, Criteria 2, 4 and 5.

The IC steam supply pipes include flow restrictors, and the IC condensate drain pipes are of limited area so that an IC piping or tube rupture in the safety-related ICS/PCC pool will limit flow-induced dynamic loads and pressure buildup in the ICS/PCC pool. ~~Also, guard pipes and special transition fittings~~ Penetration sleeves are used at the locations where the IC steam supply and condensate return pipes enter the pool at the containment pressure boundary. These penetration sleeves are designed and constructed in accordance with the requirements specified in Subsection 3.6.2.1.1. ~~The design intent for these guard pipes and transitions is either to show that the stresses and fatigue usage do not exceed special limits established by SRP 3.6.2 Branch Technical Position MEB 3-1, or to show that the guard pipes and transition fittings do not experience crack initiation or crack growth by proof testing. Proof testing would be done by using experimental stress analysis cyclic tests in accordance with ASME Code Section III, Appendix II, Subarticle II-1500, except that crack initiation or excessive crack growth will be used as the acceptance basis instead of full crack growth causing a leak, and the number of test cycles will be based on the periodic ISI inspection interval for these parts~~.

The ICS valve actuators are to be qualified for service inside the drywell for continuous service under normal conditions and to be operable for 4 hours with a steam environment. Thereafter, the valves are required to remain in their last position.

The ICS steam supply lines, condensate return lines, instrument lines, and vent lines that penetrate containment are provided with isolation valves to satisfy containment isolation requirements as discussed in Subsections 6.2.4 and 7.3.3.

Compliance of instrumentation and control equipment is addressed in Subsection 7.4.4.3.

5.4.6.4 Testing and Inspection Requirements

Inspection

During plant outages, routine ISI is required for the isolation condenser, piping containment penetration sleeves, and supports according to ASME Code Section III and Section XI (requirements for design and accessibility of welds).

IC removal for routine inspection is not required.

Ultrasonic inspection is required for IC tubes/headers welds. IC tubes will be inspected by the eddy current method.

Testing

Periodic heat removal capability testing of the ICs is required during plant operation. This test is accomplished using the temperature recorder located downstream of the isolation valve F004, together with the differential pressure recorder which gets the signal form one of the dPTs, on the condensate return line.

During normal plant operation, a periodic surveillance test of normally-closed valves F005 and F006 on condensate line to RPV, being moved into an open condition, will be performed.

The following paragraphs summarize the basis for SBWR compliance with the requirements imposed by Criterion 55.

6.2.4.3.2.1 Influent Lines

Influent lines, which penetrate the containment directly to the RCPB, are equipped with at least two isolation valves, one inside the containment and the other as close to the external side of the containment as practical. Table 6.2-13 lists the influent pipes that comprise the RCPB and penetrate the containment. The table summarizes the design of each line as it satisfies the requirements imposed by General Design Criterion 55.

Feedwater Line

The feedwater line is part of the reactor coolant pressure boundary as it penetrates the drywell to connect with the reactor pressure vessel. It has two automatically closing isolation valves. The isolation valve inside the containment is a check valve, located as close as practicable to the containment wall. Outside the containment is a spring-check valve located as close as practicable to the containment wall. The spring-check valve outside containment is provided with an air-opening, spring-closing operator which, upon remote manual signal from the main control room, provides additional seating force on the valve disc to assist in long-term leakage protection. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation.

Isolation Condenser Condensate and Venting Lines

The isolation condenser condensate lines penetrate the containment and connect directly to the reactor pressure vessel. The isolation condenser venting lines extend from the isolation condenser through the containment and connect together downstream of two normally closed control valves in series. The venting line terminates below the minimum drawdown level in the suppression pool. Each IC condensate line has two open isolation gate-valves located in the containment where they are protected from outside environmental conditions which may be caused by a failure outside the containment. In case of the venting lines there are two normally closed control globe-valves in series with isolation globe-valves. The condensate lines are automatically isolated when leakage is detected.

The IC isolation valves and the pipes penetrating the containment are designed in accordance to ASME Code Section III, Class 1 Quality Group A, Seismic Category I. Penetration sleeves used at the locations where the condensate return pipes exit the pool at the containment pressure boundary, are designed and constructed in accordance with the requirements specified in Subsection 3.6.2.1.1. ~~Also, guard pipes and special transition fittings are used between the containment penetration and the outer valve. The design intent for these guard pipes and transitions is to show that they meet the intent of BTP MEB 3-1 by analysis or proof testing.~~ In addition the IC system force is capable of closing an isolation valve. Refer to Subsection 5.4.5 for Main Steamline Isolation System description.

Isolation Condenser Steam Supply Lines

The isolation condenser steam supply lines penetrate the containment and connect directly to the reactor pressure vessel. Two isolation gate-valves are located in the containment where they are protected from outside environmental conditions which may be caused by a failure outside the containment. The isolation valves in each IC loop are signaled to close automatically on excessive flow. The flow is sensed by four differential flow transmitters in either the steam supply line or the condensate drain line. The isolation valves are also automatically closed on high radiation in the steam leaving an IC-pool compartment. The isolation functions are based on any 2 out of 4 channel trips.

The IC isolation valves and the pipe penetrating the containment are designed in accordance to ASME Code Section III, Class 1 Quality Group A, Seismic Category I. Penetration sleeves used at the locations where the IC steam supply lines enter the pool at the containment pressure boundary are designed and constructed in accordance with the requirements specified in Subsection 3.6.2.1.1. ~~Also, guard pipes and special transition fittings are used between the containment penetration and the outer valve. The design intent for these guard pipes and transitions is to show that they meet the intent of BTP MEB 3-1 by analysis or proof testing.~~ In addition to the IC isolation valves, the IC system outside the containment consists of a closed loop designed to ASME Code Section III, Class 2, Quality Group B, Seismic Category I, which is a "passive" substitute for an open "active" valve outside the containment. This closed loop substitute for an open isolation valve outside the containment implicitly provides greater safety. The combination of an already isolated loop outside the containment plus the series automatic isolation valves inside the containment comply with the intent of isolation provisions of US NRC Code of Federal Regulations 10CFR50, Appendix A, Criterion 55 and 56.

Reactor Water Cleanup System /Shutdown Cooling System

The Reactor Water Cleanup/Shutdown Cooling System(RWCU/SDC) takes its suction from the reactor pressure vessel. The RWCU/SDC suction lines of each loop are isolated by one automatic nitrogen operated gate valve inside and two parallel motor operated gate valves outside the containment. During normal operation the larger of these parallel valves (used for shut down cooling) is closed.

RWCU/SDC pumps, heat exchangers and demineralizers are located outside the drywell.

RAI Number: ECGB.12

Question:

In SSAR Section 3.7.3, it is stated that for seismic subsystem analysis of ASME components, ASME Section III, Appendix N, "Dynamic Analysis Methods," is applicable. Appendix N is a non-mandatory appendix that is still evolving and does not currently agree with some staff positions. Therefore, it has not been endorsed by the staff, and the staff has no immediate plans to review it. Some of the rules which are either in Appendix N, or are being proposed for future addenda to that standard, and which do not agree with staff positions, address issues such as damping values, use of the load coefficient method, use of the independent support motion response spectrum method of analysis, and the 50- percent nonexceedance probability level in N-1723.2, N-1724, and N-1725 of Appendix N. Revise the SSAR to delete all references to Appendix N and replace them with applicable RGs, SRPs, or staff approved ASME Code Cases.

GE Response:

The ASME Boiler and Pressure Vessel Committee meets to consider revisions of the rules as dictated by technological development. As part of this process, ASME Section III, Appendix N has evolved, in part, to address the overly conservative seismic design requirements that exist in many regulations, and which has contributed to the overall lack of safety margins for the plant. A good example of this excess conservatism is found in the seismic design of piping systems, which has resulted in inflexible piping with very high stresses for normal operating conditions this, in turn, contributes to ancillary problems such as stress corrosion cracking and high fatigue usage. Hence, the use of a more realistic, but still conservative, approach as defined by the ASME Section III, Appendix N, "Dynamic Analysis Methods," will ensure improved overall plant safety for the SBWR.

For the reasons outlined above, it is the intention to keep the present reference to Appendix N, and to request NRC approval based upon the advances made in the technology of understanding seismic behavior of structures, piping, and equipment.

RAI Number: ECGB.13

Question:

In SSAR Subsection 3.7.3.1, Electric Power Research Institute (EPRI) NP-6628 (NCIG-14), "Procedure for Seismic Evaluation and Design of Small Bore Piping," is referenced as an alternate procedure to be used in lieu of seismic analysis for piping 2 inches and smaller in diameter. This procedure incorporates, in part, the use of a seismic experience-based approach for the design or qualification of small bore safety-related piping. The staff has not accepted this procedure. Currently, the staff only accepts a suitable dynamic analysis or a suitable qualification test except when the use of an equivalent static analysis has been demonstrated to be adequate for the design of such piping systems. Revise Subsections 3.7.3.1 and 3.7.6 to delete the reference to EPRI NP-6628 (NCIG-14).

GE Response:

NCIG-14 is an analytical approach to the design of small bore piping, and is based upon the results of test programs. The NRC published NUREG-1061, Volume 2, which concluded that piping installed in non-nuclear facilities performed extremely well in strong motion earthquake, and recommended that some of the ultra-conservatism in design be reduced. After the NRC staff approves NCIG-14, Subsections 3.7.3.1 and 3.7.6 will be revised to show that NCIG-14 will be used.

RAI Number: ECGB.15

Question:

During the July 17, 1991, meeting, the staff was also informed that for predicting response of the SBWR internals to loss-of-coolant accident (LOCA) using the GDCS, a 1/508 sector-scaled SBWR test was performed, such that the data base can be used to qualify the thermal-hydraulic computer codes (TRAC and TRACG) for accident analysis. However, for normal operation, since the flow rate in the SBWR core area is dependent upon natural circulation, the flow velocities may vary in a broad range under different operating conditions. In cases of low flow rate, thermal mixing inside the reactor vessel may not be thorough, and the flow may be stratified into several regions with different thermal conditions. In such cases, reactor internal components may experience uneven thermal loads and result in high thermal stresses and high cumulative fatigue effects. Since the thermal loads are difficult to be accurately predicted analytically due to complexity of flow-pass geometries and complicated boundary conditions, an instrumented full-scale prototype testing of reactor internals under various operating transients appears necessary for confirming the thermal loads for the reactor internal component design. Discuss if such a test is planned, or if none is planned, why it is not necessary.

GE Response:

No full scale measurement of thermal loads due to thermal stratification is necessary, because as discussed in the response to SRXB.32, the SBWR operation will be controlled such that stratification is dissipated by the reactor water cleanup system at low core flow. Therefore a bounding temperature difference can be defined for use in a conventional thermal stress analysis.

RAI Number: ECGB.17

Question:

In SSAR Subsection 3.9.3.7.1, it is stated that to minimize the use of snubbers, special engineered pipe supports such as energy absorbers and limit stops may be used.

a) With respect to energy absorbers, it should be noted in this subsection that (1) ASME Code Case N-420 can only be used as conditioned by RG 1.84, and (2) ASME Code Case N-420 cannot be used in the same analysis that uses the damping values in ASME Code Case N-411. Revise Subsection 3.9.3.7.1 and any other applicable subsection in the SSAR to add these conditions.

b) The use of limit stops is currently being reviewed by the staff on a plant-specific basis. One plant has been conditionally approved to use this alternative in a part of one pilot piping system. Pending the results of the staff's evaluation of this program, the use of limit stops is not acceptable. Revise Subsection 3.9.3.7.1 and any other applicable subsection to either delete the paragraph on limit stops or commit to using this alternative only after it has been approved by the staff.

GE Response:

Sections 3.7.1.2 and 3.9.3.7.1 of the SSAR will be revised and Section 3.7.3.3.3 of the SSAR will be added in Amendment 1 (see attached).

a) Section 3.9.3.7.1 has been revised to state that Code Case N-420 can only be used if the information required by Regulatory Guide 1.84 is provided to the regulatory agency. In addition, Section 3.9.3.7.1 has been revised so that it references a new Section 3.7.3.3.3 "Modeling of Special Engineered Pipe Supports." This new section provides the analytical requirements. Section 3.7.1.2 has been revised to state that ASME Code Case N-411-1 damping cannot be used for analyzing linear energy absorbing supports designed in accordance with ASME Code Case N-420.

b) The new Section 3.7.3.3.3, referenced by subsection 3.9.3.7.1, specifies that if these special devices are used, the modeling and analytical methodology will be in accordance with methodology accepted by the regulatory agency at the time of certification or at the time of application.

The time histories of the two horizontal components also satisfy the Power Spectra Density (PSD) requirement stipulated in Appendix A to SRP 3.7.1. The computed PSD functions envelop the target PSD of a maximum 0.3*g* acceleration with a wide margin in the frequency range of 0.3 Hz to 24 Hz as shown in Figure 3.7-18 and Figure 3.7-19 for the H1 and H2 components, respectively. In these figures the curve labeled as 80% of the target PSD is the minimum PSD requirement.

The time histories of three spatial components are checked for statistically independency. The cross-correlation coefficient at zero time lag is 0.01351 between H1 and H2, 0.07037 between H1 and VT, and 0.07367 between H2 and VT. All of them are less than 0.16 as recommended in the reference of RG 1.92. Thus, H1, H2, and VT acceleration time histories are mutually statistically independent.

3.7.1.2 Percentage of Critical Damping Values

Damping values of various structures and components are shown in Table 3.7-1 for SSE dynamic analysis. These damping values are consistent with RG 1.61 SSE damping. For ASME Section III, Division 1 Class 1, 2, and 3, and ASME/ANSI B31.1 piping systems, damping values of ASME Code Case N-411-1 may be used as permitted by RG 1.84, in place of RG 1.61 damping. ASME Code Case N-411-1 damping can not be used for analyzing linear energy absorbing supports designed in accordance with ASME Code Case N-420. The damping values shown in Table 3.7-1 are applicable to all modes of a structure or component constructed of the same material. Damping values for systems composed of subsystems with different damping properties are obtained from the procedures described in Subsection 3.7.2.13.

3.7.1.3 Supporting Media for Category I Structures

The Se... Category I structures have concrete mat foundations supported on soil, rock or compacted backfill. The embedment depth, dimensions of the structural foundation, and total structural height for each structure are given in Subsection 3.8.5.1. The soil conditions considered for the design of the standard plant are described in Appendix 3A.

3.7.2 Seismic System Analysis

This section applies to building structures that constitute primary structural systems. The reactor pressure vessel (RPV) is not a primary structural component but, due to its strong dynamic interaction with supporting structure, is considered as part of the primary system of the reactor building for the purpose of dynamic analysis.

3.7.2.1 Seismic Analysis Methods

Analysis can be performed using any of the following methods:

- time history method;

locations of large masses, such as valves, pumps, and motors, and with locations of significant geometry change. All concentrated weights on the piping systems, such as the valves, pumps, and motors, are modeled as lumped mass rigid systems if their fundamental frequencies are greater than the cutoff frequency in Subsection 3.7.2.1.1. On straight runs, mass points are located at spacings no greater than the span which would have a fundamental frequency equal to the cutoff frequency stipulated in Subsection 3.7.2.1.1 when calculated as a simply supported beam with uniformly distributed mass. The torsional effects of valve operators and other equipment with offset center of gravity with respect to the piping center line are included in the analytical model. Furthermore, all pipe guides and snubbers are modeled so as to produce representative stiffness. The equivalent linear stiffness of the snubbers is based on actual dynamic tests performed on prototype snubber assemblies or on data provided by the vendor. The stiffness of the supporting structures is included in the analysis, unless the supporting structure can be shown to be rigid.

3.7.3.3.2 Equipment

For dynamic analysis, equipment is represented by lumped-mass system which consists of discrete masses connected by massless elements. The criteria used to lump masses are as follows:

- The number of modes of a dynamic system is controlled by the number of masses used; therefore, the number of masses is chosen so that all significant modes are included. The number of masses or dynamic degrees of freedom is considered adequate when additional degrees of freedom do not result in more than a 10% increase in response. Alternatively, the number of dynamic degrees of freedom is no less than twice the number of modes below the cutoff frequency of Subsection 3.7.2.1.1.

- Mass is lumped at any point where a significant concentrated weight is located. Examples are the motor in the analysis of pump stand, and the impeller in the analysis of pump shaft.

- If the equipment has free-end overhang span whose flexibility is significant compared to the center span, a mass is lumped at the overhang span.

- In the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the most critical resonant frequency of the system to ensure conservative responses under applicable floor response spectra.

3.7.3.3.3 Modeling of Special Engineered Pipe Supports

Modifications to the normal linear-elastic piping analysis methodology used with conventional pipe supports are required to calculate the loads acting on the supports and on the piping components when the special engineered supports, described in Subsection 3.9.3.7.1 (6), are used. These modifications are needed to account for greater damping of the energy absorbers and the non-linear behavior of the limit stops. If these special devices are used, the modeling and analytical methodology will be in accordance with methodology accepted by the regulatory agency at the time of certification or at the time of application, per the discretion of the applicant. In addition, the information required by Regulatory Guide 1.84 will be provided to the regulatory agency.

3.7.3.4 Basis for Selection of Frequencies

Where practical, in order to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are less than 1/2 or more than twice the dominant frequencies of the support structure. Moreover, in any case, the equipment is analyzed and/or tested to demonstrate that it is adequately designed for the applicable loads considering both its fundamental frequency and the forcing frequency of the applicable support structure.

3.7.3.5 Analysis Procedure for Damping

Damping values for equipment and piping are shown in Table 3.7-1 and they are consistent with RG 1.61. For ASME Section III, Division 1 Class 1, 2, and 3, and ASME/ANSI B31.1 piping systems, damping values of ASME Code Case N-411-1 may be used as permitted by RG 1.84. For systems made of subsystems with different damping properties, the analysis procedures described in Subsection 3.7.2.1.3 are applicable.

3.7.3.6 Three Components of Earthquake Motion

The applicable methods of spatial combination of responses due to each of the three input motion components are described in Subsection 3.7.2.6.

3.7.3.7 Combination of Modal Responses

The applicable methods of modal response combination are described in Subsection 3.7.2.7.

3.7.3.8 Interaction of Other Systems with Seismic Category I Systems

Each non-Category I (i.e. C-II or NS) system is designed to be isolated from any Seismic Category I system by either a constraint or barrier, or is remotely located with regard to the Seismic Category I system. If it is not feasible or practical to isolate the Seismic Category I system, adjacent non-Category I systems are analyzed according to the same seismic criteria as applicable to the Seismic Category I systems. For non-Category I systems attached to Seismic Category I systems, the dynamic effects of the non-Category I systems are simulated in the modeling of the Seismic Category I system. The attached non-Category I systems, up to the first anchor beyond the interface, are also designed obtained from an analysis, and are confirmed not to exceed the design loads for various operating conditions.

(5) Frame Type (Linear) Pipe Supports — Frame type pipe supports are linear supports as defined as ASME Section III, Subsection NF, Component Standard Supports. They consist of frames constructed of structural steel elements that are not attached to the pipe. They act as guides to allow axial and rotational movement of the pipe but act as rigid restraints to lateral movement in either one or two directions. Frame type pipe supports are designed in accordance with ASME Code Section III, Subsection NF-3000.

Frame type pipe supports are passive supports, requiring little maintenance and inservice inspection, and are normally used instead of struts when they are more economical or where environmental conditions are not suitable for the ball bushings at the pinned connections of struts. Similar to struts, frame type supports are not used at locations where restraint of pipe movement to thermal expansion significantly increases the secondary piping stress ranges or equipment nozzle loads.

The design loads on frame type pipe supports include those loads caused by thermal expansion, dead weight, and the inertia and anchor motion effects of all dynamic loads. As in the case of other supports, the forces on frame type supports are obtained from an analysis, which are assured not to exceed the design loads for various operating conditions.

~~Special Engineered Pipe Supports — In an effort to minimize the use and application of snubbers there may be instances where special engineered pipe supports are used where either struts or frame type supports cannot be applied. Examples of special engineered supports are Energy Absorbers, and Limit Stops.~~

(6) Special Engineered Pipe Supports — In an effort to minimize the use and application of snubbers there may be instances where special engineered pipe supports are used where either struts or frame-type supports cannot be applied. Examples of special engineered supports are Energy Absorbers, and Limit Stops.

Energy Absorbers — These are linear energy absorbing support parts designed to dissipate energy associated with dynamic pipe movements by yielding. When energy absorbers are used they will be designed to meet the requirements of ASME Section III Code Case N-420. Linear Energy Absorbing Supports for Subsection NF, Classes 1, 2, and 3 Construction, Section III, Division 1. The restrictions on location and application of struts and frame-type supports, discussed in (4) and (5) above, are also applicable to energy absorbers since energy absorbers allow thermal movement of the pipe only in its design directions.

Limit Stops — are passive seismic pipe support devices consisting of limit stops with gaps sized to allow for thermal expansion while preventing large seismic displacements. Limit stops are linear supports as defined as ASME Section III, Subsection NF, and are designed in accordance with ASME Code Section III, Subsection NF-3000. They consist of box frames constructed of structural steel elements that are not attached to the pipe. The box frames allow free movement in the axial direction but limit large displacements in the lateral direction.

Subsection 3.7.3.3.3 provides the analytical requirements for special engineered pipe supports. The information required by Regulatory Guide 1.84 shall be provided to the regulatory agency, when Code Case N-420 is used to design linear energy absorbing supports.

3.9.3.7.2 Reactor Pressure Vessel Support Skirt

The SBWR RPV support skirt is designed as an ASME Code Class 1 component support per the requirements of ASME Code Section III, Subsection NF*. The loading conditions and stress criteria are given in Tables 3.8-2 and 3.8-3, and the calculated stresses meet the Code allowable stresses at all locations for various plant operating conditions. The stress level margins assure the adequacy of the RPV support skirt. An analysis for buckling shows that the support skirt complies with Subparagraph F-1332.5 of ASME III, Appendix F, and also meet the requirements of the vessel design document as given by the criteria stated below. The permissible skirt loads at any elevation, when simultaneously applied, are limited by the following interaction equation:

$$(P/P_{crit}) + (q/q_{crit}) + (\tau/\tau_{crit})^2 < (1/S.F.) \qquad (3.9\text{-}1)$$

where:

q = longitudinal load

P = external pressure

* Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1, 2, 3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design," analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual'" Publication T114-2/83.

RAI Number: ECGB.18

Question:

The information in SSAR Section 3.9.6 infers that exemptions from the Code testing requirements may be requested.

All of the plants which have been licensed by NRC have been permitted to request relief from the ASME Section XI inservice testing (IST) rules for pumps and valves. These pumps and valves are generally installed in systems in which it is impractical to meet the Section XI rules because of limitations in the system design which preclude testing without significant design changes. In other cases, the staff approved alternatives to the Section XI requirements because imposition of the Section XI rules would have resulted in hardships to the licensee without a compensating increase in the level of safety. The underlying reason for the regulation allowing these reliefs from the code was that the detailed system designs for all of these plants were essentially completed prior to the time that the staff promulgated 10 CFR 50.55a(g) that incorporated by reference the ASME Code Section XI rules. A plant such as SBWR, for which the final design is not complete, has sufficient lead time available to include provisions for this type of testing in the detailed design of applicable piping systems. Therefore, exemptions from the applicable code testing requirements will not be granted for SBWR. However, with regard to subsequent or future code revisions to the applicable ASME Code for the SBWR plant, requests for relief from certain updated code requirements may still be submitted for staff review in accordance with 10 CFR 50.55a(f). Revise SSAR Subsection 3.9.6 to provide a more explicit commitment that SBWR will be designed to accommodate testing per the code requirements for IST of valves (and pumps, if applicable).

GE Response:

Table 3.9-8 submitted to the NRC on February 28, 1993, provides details of the inservice testing program for the SBWR. This section references ASME OM Code-1990, rather than ASME Section XI. The ASME Board on Nuclear Codes and Standards recognized that O&M is the appropriate committee to establish inservice testing requirements and voted to proceed with making the O&M Standard stand on its own, with the objective of the eventual deletion of IST from Section XI. The NRC blanket statement position that "exemptions from the applicable code testing requirements will not be granted for SBWR" is considered unreasonable. It is inconceivable that in the entire IST scope there would not be at least one valid reason to have an exception to the code. For example, if testing a valve would lead to depressurizing the reactor vessel, there might be a reason to have an exemption and delay the valve's IST until a refueling outage. The exemptions currently referred to in Section 3.9.6 are those allowed by OM Code rules. It is expected that upon review of the proposed IST program, the NRC will concur with the statements made in Section 3.9.6.

RAI Number: ECGB.19

Question:

In SSAR Section 3.9.6, GE stated that safety-related pumps and valves will be included in the IST program for the SBWR. The unique SBWR design places significant reliance on passive safety systems, but also depend on non-safety systems (which are traditional safety systems in current LWRs) to prevent challenges to passive systems. Therefore, it is very important that testability of both safety-related valves and important non-safety pumps and valves be provided early in the design phase. The applicant is requested to provide detailed information to ensure that all safety-related valves can be in situ tested to demonstrate their design capabilities and to monitor their condition.

The staff has not completed it's review of the extent to which important non-safety components may have to meet safety-grade criteria. However, there are uncertainties concerning the lack of a proven operational performance history for the valves in the passive systems. These uncertainties may increase the need to rely on the important non-safety systems and components in providing the defense-in-depth to prevent and mitigate accidents and core damage. The staff is still evaluating this issue for the passive plant designs. The specific staff positions on the inservice testing requirements for the important non-safety components will be determined when the staff completes its review of the issue of regulatory treatment of non-safety systems. The applicant will then be requested to revise Section 3.9.6 to agree with the staff's position.

GE Response:

The SSAR describes the general plan for the inservice testing (IST) program for safety-related systems in Sections 3.9.6, 3.9.6.1 and 3.9.7.3. It is not practical to undertake the enormous expense of designing all pumps and valves in non-safety systems to be IST testable, just because the NRC has not yet agreed on criteria for which plant components should be tested. The ongoing industry program and discussions with the NRC on the regulatory treatment of non-safety systems will resolve this issue for non-safety systems. SSAR Section 3.9 will be revised in a future amendment to reflect the final industry resolution of this issue as it applies to inservice testing of important non-safety components.

RAI Number: ECGB.20

Question:

In SSAR Section 3.9.6.2 of the SSAR, GE stated that the motor operated valves (MOVs) equipment specifications require the incorporation of the results of either in situ or prototype testing with full flow and pressure and/or full differential pressure to verify the proper sizing and correct switch settings of the valves. In Section 3.9.7.3 of the SSAR, GE also stated that the concerns and issues identified in Generic Letter (GL) 89-10 for MOVs will be addressed by the applicant referencing the SBWR design before plant startup. The method of assessing the loads, the method of sizing the actuator, and the setting of torque and limit switches will be specifically addressed. However, the staff has determined that all the concerns and issues identified in GL 89-10 and its supplements that relate to tests, analyses, and acceptance criteria to determine the adequacy of valve design and to ensure the ability of MOVs to meet functional performance requirements under all design basis conditions, including recovery from inadvertent valve mispositioning, must be addressed to demonstrate the design basis capability of MOVs. The staff has also determined that this issue should be addressed under a generic inspections, tests, analyses, and acceptance criteria (ITAAC) rather than a combined license (COL) action item. GE should develop an acceptable generic ITAAC for demonstrating MOV capability, as discussed above.

GE Response:

The generic ITAAC for demonstrating MOV capability will be included in the applicable sections in Amendment 1 to the Tier 1 Design Certification Document. The revised ITAAC sections will be consistent will the ITAAC for demonstrating MOV capability approved by the NRC for ABWR.

RAI Number: ECGB.21

Question:

In the SSAR, GE has committed that the MOV equipment specifications will require the incorporation of the results of either in situ or prototype testing with full flow and differential pressure to verify the proper sizing and switch settings of the valves. GE also committed that all SBWR safety-related piping systems will incorporate provisions for testing to demonstrate the operability of check valves under design basis conditions.

Based on operating experience, the staff has determined that a similar commitment is needed for the specifications for other power-operated valves to incorporate the results of either in plant or prototype testing to verify design basis capability. Based on past experience with estimating thrust and torque requirements and other parameters for valve operation, the staff believes that this assurance cannot be provided by analytical approaches alone and will require that proper sizing and adjustment of other power-operated valves be verified by a generic ITAAC. GE should develop an acceptable generic ITAAC for demonstrating the capability of other power-operated valves.

GE Revised Response:

The SBWR SSAR submittal dated February 28, 1993 included ITAACs for safety-related valves based on discussions with the NRC staff to develop the ABWR ITAACs. The SBWR ITAACs are intended to follow the agreements reached in these discussions where applicable and will be revised as further agreements are reached on the scope and methods for demonstrating the capability of safety-related valves.

RAI Number: ECGB.22

Question:

Several piping systems connected to the reactor coolant pressure boundary have design pressures below the rated reactor coolant system (RCS) pressure. Also some systems that are rated at full reactor pressure on the discharge side of pumps have pump suction pressure below RCS pressure. To protect these systems or portions of systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high-pressure RCS and the low-pressure system. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems.

In SSAR Section 3.9.6, GE stated that the periodic leak testing of the RCS pressure isolation valves (PIV) in Table 3.9.8 will be performed in accordance with Chapter 16 surveillance requirement (SR) 3.6.1.5.10. The referenced SR appears to be incorrectly identified and the correct section should be SR 3.4.3.1. SR 3.4.3.1 states that the RCS PIV leak testing frequency will be in accordance with inservice testing program or once per refueling interval.

However, it should be noted that the above-referenced inservice testing program (SSAR Table 3.9.8) will not be submitted by GE until February 28, 1993. Therefore, the staff's review of this issue cannot be completed at this time. GE is requested to provide a list of RCS PIVs. Moreover, the staff has determined that the leak testing frequency as stated in SR 3.4.3.1 is not fully acceptable for SBWR. GE is requested to address other leak testing frequencies that are contained in several of the standard TS and currently implemented by many operating plants. Those frequencies include leak testing prior to entering Mode 2 whenever the unit has been in Mode 5 for 7 days or more, if leak testing has not been performed in the previous 9 months, and leak testing within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

GE Response:

A list of RCS PIVs is included in Table 3.9-8 Inservice Testing.

The leak testing frequencies as referenced in SR 3.4.3.1 are the same as those in NUREG 1434, Rev. 0 which has been the basis for these Technical Specifications. The requirement to perform leak testing at the frequencies addressed in this RAI first appeared in an NRC letter to all LWR licensees, dated February 23, 1980. The requirements resulted from the WASH-1400 Study, however, this study concluded that acceptable methods to assure component integrity not only included performing leak tests at these two frequencies, but also included continuous pressure monitoring on the low pressure side of each check valve. The current SBWR design includes this continuous pressure monitoring and current Technical Specifications were not modified to require these two frequencies. Thus, as specified in the NRC letter, continuous pressure monitoring on the low pressure side of the susceptible check valves are part of current plant design and do not need to be included in the Improved Technical Specifications.

SSAR Section 3.9.6 will be revised in Amendment 1 (see attached) to state that periodic leak testing will be performed in accordance with Chapter 16 SR 3.4.3.1.

requirements for safety-related valves including those listed in the technical specifications (Chapter 16) and the containment isolation system (Subsection 6.2.4.) For example, the periodic leak testing of the reactor coolant pressure isolation valves in ~~Table 3.8-9~~Table 3.9-8 will be performed in accordance with Chapter 16 Surveillance Requirement ~~SR3.6.1.5.10~~SR3.4.3.1. This plan will include baseline pre-service testing to support the periodic inservice testing of the components. Depending on the test results, the plan will provide a commitment to disassemble and inspect the safety-related valves when the OM Code limits are exceeded, as described in the following paragraphs. The primary elements of this plan, including the requirements of Generic Letter 89-10 for motor operated valves, are delineated in the subsections to follow. (Refer to Subsection 3.9.7.3 for COL license information requirements.)

3.9.6.1 Inservice Testing of Safety-Related Valves

Check Valves

All SBWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design conditions. Inservice testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The Subsection ISTC tests will be performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.8-9. A program will be developed by the applicant referencing the SBWR design to establish the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (Refer to Subsection 3.9.7.3(1) for COL license information requirements.)

Motor Operated Valves

The motor operated valve (MOV) equipment specifications require the incorporation of the results of either in-situ or prototype testing with full flow and pressure or full differential pressure to verify the proper sizing and correct switch settings of the valves. Guidelines to justify prototype testing are contained in Generic Letter 89-10, Supplement 1, Questions 22 and 24 through 28. The applicant referencing the SBWR design will provide a study to determine the optimal frequency for valve stroking during inservice testing such that unnecessary testing and damage is not done to the valve as a result of the testing. (Refer to Subsection 3.9.7.3(1) for COL license information requirements).

The concerns and issues identified in Generic Letter 89-10 for MOVs will be addressed prior to plant startup. The method of assessing the loads, the method of sizing the actuators, and the setting of the torque and limit switches, will be specifically addressed. (Refer to Subsection 3.9.7.3(1) for COL license information requirements.)

RAI Number: ECGB.23

Question:

In SSAR Section 3.9.6, GE stated that IST of safety-related pumps and valves will be performed in accordance with the requirements of ASME OM Code 1990, Subsections ISTB, ISTC, and Appendix I. It should be noted that Subsections ISTB and ISTC of the ASME OM Code 1990 are essentially the same as OM Standards Part 6, "Inservice Testing of Pumps," and Part 10, "Inservice Testing of Valves," respectively. However, OM Standards Part 6 and Part 10 are referenced in Section XI of the 1988 Addenda and 1989 Edition. The 1988 Addenda and the 1989 Edition of Section XI have been incorporated by reference into 10 CFR 50.55a and are acceptable for the passive LWR IST provided the analysis of leakage rates and corrective action requirements of Paragraphs 4.2.2.3(e) and 4.2.2.3(f) of Part 10 are applied to containment isolation valve testing. Therefore, Section 3.9.6 should be revised to refer the 1988 Addenda and 1989 Edition.

GE Response:

The ASME Board on Nuclear Codes and Standards recognized that OM is the appropriate committee to establish inservice testing (IST) requirements, with the objective of eventual deletion of IST from Section XI of the ASME Boiler and Pressure Vessel Code. As correctly identified by the NRC, the relevant sections of both the ASME OM Code 1990 and Section XI of the 1988 Addenda and the 1989 Edition of the ASME Code are the same.

As noted in the February 1992 issue of the SSAR, the IST program plan is based on ASME OM Code 1990, Subsections ISTB and ISTC and Appendix I. Containment isolation valve testing is covered by Subsection ISTC, Paragraph 4.3.2 of ASME OM Code 1990 and by Section 6.2.6.3 of the SSAR, and is controlled by surveillance Requirement SR 3.6.1.3.7 of the Technical Specifications. SR 3.6.1.3.7 limits containment isolation valve combined leakage to a total of 0.227 m3/hr (1 gpm) times the total number of CIVs hydro-statically tested lines that penetrate the containment when the isolation valves are tested at 1.1 times the peak calculated containment pressure. Therefore, the additional leakage rate requirements specified in Paragraphs 4.2.2.3(e) and 4.2.2.3 (f) of Part 10 of the 1988 Edition of the OM Standard are not required.

RAI Number: ECGB.24

Question:

Regarding SSAR Section 3.7, "Seismic Design," the seismic Category I systems and components are designed to remain functional for earthquake loadings. Provide the basis for why this section of the SSAR does not address the structural integrity of the systems and components.

GE Response:

Section 3.7 of the SSAR will be revised in Amendment 1 to state that Seismic Category I structures, systems and components (SCC) are designed to remain functional during and subsequent to a design basis earthquake (see attached). Therefore, all Seismic Category I SCC are designed to retain their structural integrity as necessary to perform their intended function(s).

3.7 Seismic Design

For seismic design purposes, all structures, systems, and components of the Simplified Boiling Water Reactor (SBWR) standard plant are classified into Seismic Category I (C-I), Seismic Category II (C-II), or Non-Seismic (NS) in accordance with the requirements to withstand the effects of the Safe Shutdown Earthquake (SSE) as defined in Section 3.2. For those C-I and C-II structures, systems, and components in the reactor building complex, the effects of other dynamic loads caused by reactor building vibration (RBV) caused by suppression pool dynamics are also considered in the design. Although this section addresses seismic aspects of design and analysis in accordance with Regulatory Guide 1.70, the methods of this section are also applicable to RBV dynamic loadings, unless noted otherwise.

The safe shutdown earthquake (SSE) is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which Seismic Category I structures, systems, ~~systems~~ and components are designed to remain functional. These systems and components are those necessary to ensure the following:

- the integrity of the reactor coolant pressure boundary;
- the capability to shut down the reactor and maintain it in a safe condition; and
- the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guideline exposures of 10CFR100.

Seismic Category II (C-II) includes all plant structures, systems, and components which perform no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room. Thus, this category includes the structures, systems and components whose structural integrity, not their operational performance, is required. Seismic Category II structures, systems, and components are designed and/or so physically arranged so that the SSE would not cause unacceptable structural interaction or failure. For fluid systems, this requires an appropriate level of pressure boundary integrity when located near sensitive equipment. Appropriate seismic ductility factors are selected for design to take credit for realistic amounts of energy dissipation in C-II items. Seismic Category II (C-II) items are those corresponding to positions C.2 and C.4 of Regulatory Guide 1.29.

Non-seismic structures and equipment are those which do not fall into Seismic Category I or II definitions. NS structures and equipment are designed for seismic requirements in accordance with the Uniform Building Code for Zone 2A. The building structures

RAI Number: ECGB.25

Question:

SSAR Section 3.7 states that the exhaust stack is classified as non-safety-related. Provide the basis for how postulated failures of this structure would not affect the function or integrity of any safety-related component or structure.

GE Response:

See Response to ECGB-9.

RAI Number: ECGB.26

Question:

Regarding Figures 3.7.1 and 3.7.2 in SSAR Section 3.7.1.1.2, "Design Time History," show the design response spectra for damping ratios of 2, 5, 7, and 10 percent. However, Figures 3.7.6 through 3.7.17 show the spectra enveloping for damping ratios of 2, 3, 4, and 7 percent. What is the basis for not showing that Figures 3.7.1 and 3.7.2 should reflect 3 and 4 percent damping response spectra values and 5- and 10-percent damping ratios for Figures 3.7.6 through 3.7.17? The staff believes it is not acceptable for GE to use the 5- and 10-percent damping ratios in the analysis and design of structures, systems, and components, if the design time history cannot satisfy the enveloping criteria for these two damping ratios. Also, please show (or provide the basis for not including) the power spectrum density function enveloping condition for the vertical time history.

GE Response:

1) Figures 3.7.1 and 3.7.2 will be revised in Amendment 1 of the SSAR to show damping values of 2%, 3%, 4%, 5%, and 7%.

2) The SRP does not require power spectral densities (PSD) for vertical accelerations; however, these will be provided in Amendment 1 of the SSAR.

RAI Number: ECGB.27

Question:

Regarding SSAR Section 3.7.1.2, "Percentage of Critical Damping Values," what is the basis for not showing or listing the damping values for the electrical components such as cable trays, conduit, heating, ventilation, and air conditioning, etc?

GE Response:

SBWR will follow the industrial practice and the development being conducted by ASCE, ASME and EPRI for a set of damping values for cable trays, conduit, and HVAC duct.

RAI Number: ECGB.28

Question:

Regarding SSAR Section 3.7.2.1.1, "Time History Method," what is the definition for the term "highest frequency (or shortest period) of significant," and why is this not defined in the SSAR?

GE Response:

The term "Highest frequency (or shortest period)of significance" is provided in the last paragraph of Section 3.7.2.1.1 on Page 3.7-5.

RAI Number: ECGB.29

Question:

What is the basis for the following statement contained in SSAR Section 3.7.2.1.1? "For the frequency domain solution, the frequency interval is selected to accurately define the transfer functions at structural frequencies within the range of significant."

GE Response:

The SSAR Section 3.7.2.1.1 contains the following statement "For the frequency domain solution, the frequency interval is selected to accurately define the transfer functions at structural frequencies within the range of significance." The basis for this statement is contained within the Standard ASCE 4-86 "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures" dated September 1986. The section of ASCE 4-86 that the above referenced statement was obtained is Section 3.2.4(b) "The frequency interval shall be selected to accurately define the transfer functions at structural frequencies."

RAI Number: ECGB.30

Question:

Regarding SSAR Section 3.7.2.3, "Procedures Used for Analytical Modeling," what is the basis for not including in the seismic analysis the lump mass to the node points and the consideration of the dynamic effects such as water sloshing, etc.?

GE Response:

The lumped mass is included in the seismic analysis at node points, and consideration has been given to the hydrodynamic coupling effects. The fourth paragraph of Section 3.7.2.3 provides this information. The water sloshing effects on structures will also be included in the design.

RAI Number: ECGB.31

Question:

Regarding SSAR Section 3.7.2.5, "Development of Floor Response Spectra (FRS)," what is the basis for justifying the acceptability of the direct generation method of the FRS?

GE Response:

The Standard Review Plan (SRP) 3.7.2 allows the use of direct generation of floor response spectra (FRS) on a case-by-case approval. The reference 3.7.1 provides the technical basis for the acceptability of this direct generation of FRS.

RAI Number: ECGB.32

Question:

Regarding SSAR Section 3.7.2.7, "Combination of Modal Responses," please clarify why the combination methods discussed in this section are also applicable for the modal time history analysis.

GE Response:

Section 3.7.2.7 is limited to Response Spectrum Method.

RAI Number: ECGB.33

Question:

Regarding SSAR Section 3.7.2.14, "Dynamic Stability of Seismic Category I Structures," please provide the basis for not discussing the problem of dynamic instability of seismic Category I structures due to sliding.

GE Response:

The discussion of the problem of .ynamic instability of seismic Category 1 structures due to sliding is provided in Appendix 3E.7.6 "Foundation Stability."

RAI Number: ECGB.34

Question:

Regarding SSAR Section 3.7.3.12, "Seismic Category I Buried Piping, Conduits, Conduits and Tunnels," please explain the difference between the seismic Category I features and the seismic Category C-I features.

GE Response:

There is no difference between seismic Category 1 features, and the seismic Category C-I features as discussed in Section 3.7.3.12.

RAI Number: ECGB.35

Question:

Regarding SSAR Section 3.B.6, "Soil-Structure Interaction," please provide the basis for not documenting the validation and quality assurance status of the modified SASSI code which is to be used for the analysis of the SBWR structures.

GE Response:

The computer code SASSI used for the analysis of soil-structure interaction for the SBWR was validated and the quality assurance of the program and the computed results are described in Section 3b.1. The CRAY version provided to GE, identified as GE ECP SASSI01S, installed on the GE computer system, contains the same modifications and enhancements that were made to the Bechtel CRAY version, and this improved version was verified against published benchmark results. As stated earlier, all verification documentation is controlled and has been completed. For SBWR application, this version of the code was installed on Los Alamos National Lab Computer System and code revalidation was performed and solutions were found to be the same as those obtained from SASSI01S on the GE computer system.

RAI Number: ECGB.36

Question:

Regarding SSAR Section 3.7.4, "Seismic Instrumentation," Subsection 3.7.4.1 states that the number of time-history accelerographs (THAs) contained in the plant will be consistent with the number of THAs contained in draft NRC RG DG-1016 (a proposed revision to RG 1.12). Draft RG DG-1016 suggests that a plant be equipped with 8 THAs. SSAR Subsections 3.7.4.2 and 3.7.4.3 state that the plant will be equipped with 4 triaxial THAs. Please provide the basis for this apparent inconsistency.

GE Response:

The Reactor Building and Containment Structure are incorporated into one integral structure in the SBWR plant design, and the location of the four triaxial THAs meets the requirements of the draft RG-DG-1016. Duplication of THAs within the Reactor Building/Containment Structure would provide no additional information, but would greatly contribute to increased maintenance, testing and ALARA costs.

RAI Number: ECGB.37

Question:

Regarding SSAR Section 3.8, "Design of Seismic Category I Structures," please provide the basis for not including detailed structural drawing in the SSAR.

GE Response:

Detailed structural drawings were included in the February submittal of the SSAR. Please refer to Volume 15, Drawings 21.3.8-1 through 21.3.8-25.

RAI Number: ECGB.38

Question:

Regarding SSAR Section 3.8, American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690 Specifications and ASCE 4-86 have not yet been accepted by the staff. Please provide the basis for the commitment to use these standards.

GE Response:

The overly conservative seismic design requirements that exist in many regulations has contributed to the overall lack of safety margins for the plant. A good example of this excess conservatism is found in the seismic design of piping systems, which has resulted in inflexible piping with very high stresses for normal operating conditions; this, in turn, contributes to ancillary problems such as stress corrosion cracking and high fatigue usage. Hence, the use of a more realistic, but still conservative, approach as defined by the ANSI/AISC N690 and ASCE 4-86 Codes will result in increased plant safety margins.

RAI Number: ECGB.39

Question:

Regarding SSAR Section 3.8.1.7, "Design and Analysis Procedures," please provide the definition of a "critical location" and the criteria for the selection of "critical locations."

GE Response:

The definition of "critical location" as used in the SSAR Section 3.8.1.7, is that location or section in the analysis model which experiences the highest stresses or forces. There may be several different "critical locations" in a wall, beam or slab depending on the loads and loading combinations, each of which is evaluated. These evaluations determine the amount of rebar and the stresses in concrete and reinforcement for reinforced concrete elements; and determine the stresses in structural steel elements. The process of determining these critical sections is a combination of computer analysis results which show deformed shapes and stresses, engineering experience, and evaluations during the preliminary design phase. The use of this approach ensures the design is both conservative and economic.

RAI Number: ECGB.40

Question:

Regarding SSAR Section 3.8.1.7, "Design and Analysis Procedures," please provide the basis for not including detailed procedures for the reinforced concrete containment vessel (RCCV) analysis and design in the SSAR.

GE Response:

The detailed procedures referenced in the SSAR Section 3.8.1.7, "Design and Analysis Procedures" for the reinforced concrete containment vessel (RCCV) analysis and design are provided in Appendix 3E in the February 1993 submittal to the NRC.

RAI Number: ECGB.41

Question:

Regarding SSAR Section 3.8.3.1, "Description of the Internal Structures," the inner periphery radius of the diaphragm floor of 7.65m documented in the section is different from the radius of 7.8m, as shown in SSAR Figure 21.1.2-2, Sheet 2. Please explain this apparent inconsistency.

GE Response:

The inner periphery radius of the diaphragm floor is 7.65m, as documented in the SSAR Section 3.8.3.1. The dimension of 7.8m as shown in Figure 21.1.2-2 will be revised to 7.65m in Amendment 1 of the SSAR.

RAI Number: ECGB.42

Question:

Regarding SSAR Section 3.8.4.3, "Loads," please provide the basis for not addressing the flooding and tornado missile loadings in the SSAR.

GE Response:

The flooding and missile loads are addressed in the SSAR in the following sections:

Section 3.3 Wind and Tornado Loadings
Section 3.4 Water Level (Flood) Design

RAI Number: ECGB.43

Question:

Regarding SSAR Section 3.3.1, equation (3.3-1), GE states, "Importance factor I depends on the type of exposure and appropriate values of I are listed in SSAR Table 3.3-1." However, this definition of I is not consistent with that in Reference 3.3-1 which states that Importance Factor I is used to adjust the design wind speed to that with annual probabilities of being exceeded other than the value 0.02 (i.e., 50-year recurrence). This factor converts the wind speed of a 50-year recurrence to either a 25-year or 100-year recurrence and this does not depend on the type of exposure. Explain this discrepancy in applying this factor.

GE Response:

The importance factor is not a function of the type of exposure. Table 3.3-1 was incorrectly labeled. Table 3.3-1 will be revised in Amendment 1 (see attached) to indicate an importance factor of 1.00 for non-safety-related structures and an importance factor of 1.11 for safety-related structures.

Table 3.3-1 Importance Factor (I) for Wind Loads

~~Exposure C~~ Non-Safety-Related Structures	~~Exposure D~~ Safety-Related Structures
1.00	1.11

~~Notes:~~

~~1) These values of (I) are based upon Table 5 of Reference 3.3-1 but are modified to reflect the 100 year return period of the design wind velocity versus the 50 year return period basis of Reference 1.~~

~~2) Exposure categories are as defined in Section 6.5.3 of Reference 3.3-1.~~

RAI Number: ECGB.44

Question:

Regarding SSAR Table 3.3-2, this table lists the velocity pressure distribution and gust factors at various heights without providing the mean roof heights. Therefore, provide or explain the following.

- Provide the mean roof height above grade for the Reactor Building
- Explain why the windward wall pressure is 0.86Ghqh instead of 0.8Ghqz
- Provide the calculations for the values listed in this table and explain how this table is used

GE Response:

SSAR Section 3.3 will be revised in Amendment 1 of the SSAR to be consistent with the Subsection 2.3.1 wind velocity (see attached). SSAR Table 3.3-2 will be revised in Amendment 1 of the SSAR (see attached). SSAR Table 3.3-2 provides the velocity pressure distribution at various heights above grade level to enable design engineers to determine quickly and efficiently the structural effects of wind loadings for the Reactor Building.

The maximum roof height of the Reactor Building is 39.5 meters (129.3 feet) above grade. Since the roof is basically a flat roof and because using the maximum height increases the loads, the Reactor Building roof is used as the mean roof height above grade.

The windward wall pressure is a function of "qz" rather than "qh" and Table 3.3-2 has been revised accordingly (see attached).

A discussion of the development and use of Table 3.3-2 is provided in Section 3.3.1.2. Notes will be added to Table 3.3-2 in Amendment 1 of the SSAR to further explain its use (see attached).

2.0 Site Characteristics

The site characteristics information will be provided in the combined operating license (COL) applicant's Safety Analysis Report (SAR) in accordance with 10CFR52.79. (See Subsection 2.7.1 for COL applicant license information requirements.) Sections 2.1 through 2.5 of this chapter, which has the same format as Chapter 2 of NUREG-0800 standard review plan (SRP), define the limits imposed on the SRP Section II acceptance criteria by (1) the envelope of site-related parameters that the Simplified Boiling Water Reactor (SBWR) plant is designed to accommodate, and (2) the assumptions, both implicit and explicit, related to site characteristics employed in the evaluation of the SBWR design.

2.1 Geography and Demography

2.1.1 Site and Location Description

None.

2.1.2 Exclusion Area Authority and Control

None.

2.1.3 Population Distribution

None.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1-2.2.2 Identification of Potential Hazards in Site Vicinity

None.

2.2.3 Evaluation of Potential Accidents

None.

2.3 Meteorology

2.3.1 Regional Climatology

The basic speed of extreme winds used for design of structures is 49.2m/s (110 mph) at an elevation of 10m (33 feet) above grade, and it has a recurrence interval of 50 years.

3.3 Wind and Tornado Loadings

SBWR Standard Plant structures which are Seismic Category I and II are designed for tornado and extreme wind phenomena.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

Seismic Category I and II structures are designed to withstand a design wind velocity described in ~~SSAR Subsection 2.2.1~~ SSAR Subsection 2.3.1~~at an elevation of 10 m (33 feet) above grade with a recurrence interval of 100 years~~. Refer to Subsection 3.3.3 for interface requirement.

3.3.1.2 Determination of Applied Forces

The design wind velocity is converted to velocity pressure in accordance with Reference 3.3-1 using the formula:

$$qz = 0.00256Kz\,(IV)^2 \tag{3.3-1}$$

where:

Kz = the velocity pressure exposure coefficient which depends upon the type of exposure and height (z) above ground per Table 6 of Reference 3.3-1~~converted to metric units~~;

I = the importance factor which depends on the type of exposure; appropriate values of I are listed in Table 3.3-1;

V = design wind velocity mph; and

qz = velocity pressure ~~in MPa~~ (psf).

The velocity pressure (qz) distribution with height for exposure types C and D of Reference 3.3-1 are given in Table 3.3-2 for the Reactor Building. Table 3.3-3 gives correction factors to correct loads for building heights other than the 39.5 meter (129 foot) Reactor Building.

The design wind pressures and forces for buildings, components and cladding, and other structures at various heights above the ground are obtained, in accordance with Table 4 of Reference 3.3-1 by multiplying the velocity pressure by the appropriate pressure coefficients and gust factors. Gust factors are in accordance with Table 8 of Reference 3.3-1. Appropriate pressure coefficients are in accordance with Figures 2, 3a, 3b, 4, and Tables 9 and 11 through 16 of Reference 3.3-1. Reference 3.3-2 is used to obtain the effective wind pressures for cases which Reference 3.3-1 does not cover. Since the Seismic Category I and II structures are not slender or flexible, vortex-shedding analysis is not required and the above wind loading is applied as a static load.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

The design basis tornado and applicable missiles are described in ~~Subsection 2.2.1~~ Subsection 2.3.1.

Refer to Subsection 3.3.3 for COL License Information.

3.3.2.2 Determination of Forces on Structures

The procedures of transforming the tornado loading into effective loads and the distribution across the structures are in accordance with Reference 3.3-3. The procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is given in Subsection 3.5.3.1. The loading combinations of the individual tornado loading components and the load factors are in accordance with Reference 3.3-3.

The reactor building is not a vented structure. The exposed exterior roofs and walls of this structure are designed for the full pressure drop. Tornado dampers are provided on all air intake and exhaust openings. These dampers are designed to withstand the full negative pressure drop.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

All safety-related system and components are protected within tornado-resistant structures.

3.3.3 COL License Information

Site-Specific Design Basis Wind and Tornado

The site-specific design basis wind and tornado shall not exceed the wind given in ~~Subsection 2.2.1~~ Subsection 2.3.1.

Effect of Remainder of Plant Structures, Systems, and Components not Designed for Tornado Loads

All remainders of plant structures, systems, and components not designed for tornado loads shall be analyzed for the site-specific loadings to ensure that their mode of failure will not affect the ability of the Seismic Category I and II SBWR Standard Plant structures, systems, and components to perform their intended functions. (Refer to Subsection 3.3.2.3.)

3.3.4 References

3.3-1 ~~ANSI Standard A58.1~~ ASCE Standard 7-1988, Minimum Design Loads for Buildings and Other Structures, Committee A. 58.1, American National Standards Institute.

3.3-2 ASCE Paper No. 3269, Wind Forces on Structures, Transactions of the American Society of Civil Engineers, Vol. 126, Part II.

3.3-3 Bechtel Topical Report BC-TOP-3-A, Revision 3, Tornado and Extreme Wind Design Criteria for Nuclear Power Plants.

Table 3.3-1 Importance Factor (I) for Wind Loads

~~Exposure C~~ Non-Safety-Related Structures	~~Exposure D~~ Safety-Related Structures
1.00	1.11

~~Notes:~~

~~1) These values of (I) are based upon Table 5 of Reference 3.3-1 but are modified to reflect the 100 year return period of the design wind velocity versus the 50 year return period basis of Reference 1.~~

~~2) Exposure categories are as defined in Section 6.5.3 of Reference 3.3-1.~~

~~Table 3.3-2 Velocity Pressure Distribution and Gust Factors at Various Heights~~

Table 3.3-2a Design Pressure Distribution at Various Heights for Safety-Related Structures (Importance Factor = 1.11)

Height Zone z m(ft)	Windward Wall Pressure 0.8Ghqz Pa(psf)	Side Wall Suction 0.7Ghqh Pa(psf)	Roof Suction 0.7Ghqh Pa(psf)	Leeward Wall Suction 0.5Ghqh Pa(psf)
		Exposure Type C		
0-4.57 (0-15)	1544 (32)	2507 (52)	2507 (52)	1791 (37)
6.10 (20)	1641 (34)	2450 (51)	2450 (51)	1750 (37)
7.62 (25)	1727 (36)	2412 (50)	2412 (50)	1723 (36)
9.14 (30)	1805 (38)	2393 (50)	2393 (50)	1710 (36)
12.19 (40)	1906 (40)	2336 (49)	2336 (49)	1669 (35)
15.24 (50)	1999 (42)	2298 (48)	2298 (48)	1642 (34)
18.29 (60)	2088 (44)	2279 (48)	2279 (48)	1628 (34)
21.34 (70)	2157 (45)	2260 (47)	2260 (47)	1615 (34)
24.38 (80)	2225 (47)	2241 (47)	2241 (47)	1601 (33)
27.43 (90)	2292 (48)	2222 (46)	2222 (46)	1587 (33)
30.48 (100)	2340 (49)	2203 (46)	2203 (46)	1574 (33)
36.58 (120)	2438 (51)	2184 (46)	2184 (46)	1560 (33)
42.67 (140)	2533 (53)	2165 (45)	2165 (45)	1547 (32)
48.77 (160)	2610 (55)	2146 (45)	2146 (45)	1533 (32)
		Exposure Type D		
0-4.57 (0-15)	2017 (42)	2707 (57)	2707 (57)	1993 (40)
6.10 (20)	2117 (44)	2683 (56)	2683 (56)	1917 (40)
7.62 (25)	2181 (46)	2660 (56)	2660 (56)	1900 (40)
9.14 (30)	2243 (47)	2636 (55)	2636 (55)	1883 (39)
12.19 (40)	2369 (50)	2613 (55)	2613 (55)	1866 (39)
15.24 (50)	2444 (51)	2589 (54)	2589 (54)	1849 (39)
18.29 (60)	2518 (53)	2565 (54)	2565 (54)	1832 (38)
21.34 (70)	2574 (54)	2542 (53)	2542 (53)	1816 (38)
24.38 (80)	2637 (55)	2542 (53)	2542 (53)	1816 (38)
27.43 (90)	2675 (56)	2518 (53)	2518 (53)	1799 (38)
30.48 (100)	2737 (57)	2518 (53)	2518 (53)	1799 (38)
36.58 (120)	2805 (59)	2495 (52)	2429 (52)	1782 (37)
42.67 (140)	2870 (60)	2471 (52)	2491 (52)	1765 (37)
48.77 (160)	2947 (60)	2471 (52)	2471 (52)	1765 (37)

Table 3.3-2b Design Pressure Distribution at Various Heights for Non-Safety-Related Structures (Importance Factor = 1.00)

Height Zone z m(ft)	Windward Wall Pressure 0.8Ghqz Pa(psf)	Side Wall Suction 0.7Ghqh Pa(psf)	Roof Suction 0.7Ghqh Pa(psf)	Leeward Wall Suction 0.5Ghqh Pa(psf)
		Exposure Type C		
0-4.57 (0-15)	1253 (26)	2035 (43)	2035 (43)	1454 (30)
6.10 (20)	1332 (28)	1989 (42)	1989 (42)	1421 (30)
7.62 (25)	1401 (29)	1958 (41)	1958 (41)	1399 (29)
9.14 (30)	1465 (31)	1943 (41)	1943 (41)	1388 (29)
12.19 (40)	1547 (32)	1869 (40)	1869 (40)	1355 (28)
15.24 (50)	1622 (34)	1865 (39)	1865 (39)	1332 (28)
18.29 (60)	1694 (35)	1850 (39)	1850 (39)	1321 (28)
21.34 (70)	1751 (37)	1835 (38)	1835 (38)	1310 (27)
24.38 (80)	1806 (38)	1819 (38)	1819 (38)	1299 (27)
27.43 (90)	1860 (39)	1804 (38)	1804 (38)	1288 (27)
30.48 (100)	1899 (40)	1778 (37)	1778 (37)	1277 (27)
36.58 (120)	1979 (41)	1773 (37)	1773 (37)	1266 (26)
42.67 (140)	2056 (43)	1758 (37)	1758 (37)	1255 (26)
48.77 (160)	2118 (44)	1742 (36)	1742 (36)	1240 (26)
		Exposure Type D		
0-4.57 (0-15)	1637 (34)	2197 (46)	2197 (46)	1569 (33)
6.10 (20)	1718 (36)	2178 (46)	2178 (46)	1556 (33)
7.62 (25)	1770 (37)	2159 (45)	2159 (45)	1542 (32)
9.14 (30)	1821 (38)	2140 (45)	2140 (45)	1528 (32)
12.19 (40)	1923 (40)	2120 (44)	2120 (44)	1515 (32)
15.24 (50)	1984 (41)	2101 (44)	2101 (44)	1501 (31)
18.29 (60)	2043 (43)	2082 (43)	2082 (44)	1487 (31)
21.34 (70)	2089 (44)	2063 (43)	2063 (43)	1474 (31)
24.38 (80)	2140 (45)	2063 (43)	2063 (43)	1474 (31)
27.43 (90)	2171 (45)	2044 (43)	2044 (43)	1460 (31)
30.48 (100)	2222 (46)	2044 (43)	2044 (43)	1460 (31)
36.58 (120)	2276 (48)	2025 (42)	2025 (42)	1446 (30)
42.67 (140)	2330 (49)	2006 (42)	2006 (42)	1443 (30)
48.77 (160)	2392 (50)	2006 (42)	2006 (42)	1433 (30)

RAI Number: ECGB.45

Question:

Regarding SSAR Section 3.3.3, "COL license information," the site-specific design basis tornado part is missing. The site-specific design basis wind is provided in SSAR Section 2.3.1 (not 2.2.1).

GE Response:

The SSAR Section 3.3.3, COL License Information, will be revised in Amendment 1 of the SSAR (see attached) to read as follows:

<u>"Site-Specific Design Basis Wind and Tornado</u>

The site-specific design basis wind and tornado shall not exceed the wind and tornado given in Subsection 2.3.1."

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

The design basis tornado and applicable missiles are described in Subsection 2.2.1.

Refer to Subsection 3.3.3 for COL License Information.

3.3.2.2 Determination of Forces on Structures

The procedures of transforming the tornado loading into effective loads and the distribution across the structures are in accordance with Reference 3.3-3. The procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is given in Subsection 3.5.3.1. The loading combinations of the individual tornado loading components and the load factors are in accordance with Reference 3.3-3.

The reactor building is not a vented structure. The exposed exterior roofs and walls of this structure are designed for the full pressure drop. Tornado dampers are provided on all air intake and exhaust openings. These dampers are designed to withstand the full negative pressure drop.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

All safety-related system and components are protected within tornado-resistant structures.

3.3.3 COL License Information

Site-Specific Design Basis Wind and Tornado
The site-specific design basis wind shall not exceed the wind and tornado given in ~~Subsection 2.2.1~~ Subsection 2.3.1.

Effect of Remainder of Plant Structures, Systems, and Components not Designed for Tornado Loads
All remainders of plant structures, systems, and components not designed for tornado loads shall be analyzed for the site-specific loadings to ensure that their mode of failure will not affect the ability of the Seismic Category I and II SBWR Standard Plant structures, systems, and components to perform their intended functions. (Refer to Subsection 3.3.2.3.)

RAI Number: EELB.2

Question:

If not addressed in the forthcoming SSAR Section 1.8, "Interfaces for Standard Design," and Section 1.9, "Conformance with Standard Review Plan," then please provide an explanation and how the SBWR incorporates into the design the policy issues discussed in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," and the draft Commission paper dated February 27, 1992.

GE Response:

Introduction

SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, and the draft Commission paper dated February 27, 1992, identify several policy and significant technical issues pertaining to evolutionary and passive light water reactors (LWRs) and the NRC staff's recommendations concerning resolution of those issues for which the staff has completed its review. These issues are not addressed in either SSAR Section 1.8 or 1.9, but, rather, are presented herein.

The issues considered by the staff to be significant to reactor safety are listed below:

I. SECY-90-016 Issues

   A. Use of physically based source term
   B. Anticipated transients without scram (ATWS)
   C. Mid-loop operation
   D. Station blackout
   E. Fire protection
   F. Intersystem loss-of-coolant-accident
   G. Hydrogen control
   H. Core concrete interaction – ability to cool core debris
   I. High-pressure core melt ejection
   J. Containment performance
   K. Dedicated containment vent penetration
   L. Equipment survivability
   M. Elimination of operating basis earthquake (OBE)
   N. Inservice testing of pumps and valves

II. Other Evolutionary and Passive Design Issues

   A. Industry codes and standards
   B. Electrical distribution
   C. Seismic hazard curves and design parameters
   D. Leak-before-break
   E. Classification of main steamlines of boiling water reactors (BWRs)

F. Tornado design basis
G. Containment bypass
H. Containment leak rate testing
I. Post-accident sampling system
J. Level of detail
K. Prototyping
L. Inspections, tests, analyses, and acceptance criteria (ITAAC)
M. Reliability assurance program (RAP)
N. Site-specific probabilistic risk assessments
O. Severe accident mitigation design alternatives (SAMDAs)
P. Generic rulemaking related to design certification

III. Passive Design Issues Only

A. Regulatory treatment of non-safety systems
B. Definition of passive failure
C. Thermal-hydraulic stability of the simplified boiling water reactor (SBWR)
D. Safe shutdown requirements
E. Control room habitability
F. Radionuclide attenuation
G. Simplification of off-site emergency planning

The current SBWR positions on each of the above issues are contained in the paragraphs that follow, with each SBWR position preceded by a summary of each issue. Each issue is identified (in brackets following the issue headings) using the same designations as in the list above, which are consistent with the designations in the draft Commission paper dated February 27, 1992.

Use of Physically based Source Term [I.A]

Summary of Issue

As discussed in SECY-90-016, the staff's methodology for determining compliance with the siting requirements of 10CFR100 has been based on the source term provided in Technical Information Document (TID)-14844, issued in 1962. This methodology is widely acknowledged to utilize conservative assumptions.

EPRI proposed (in submittals dated October 18, 1990, and February 12, 1991) a physically based source term to be used for the licensing design basis fission product release based on a bounding severe reactor accident to be used for both the evolutionary and passive reactor designs. The EPRI-proposed source terms use release data obtained from the Severe Fuel Damage Tests at the Power Burst Facility, the LOFT source term measurements, and data from the TMI-2 post-accident examination. EPRI proposed changes in the assumptions concerning
the fission product fuel release magnitude, the fission product release timing, the chemical form of iodine, the retention of aerosol in the reactor coolant, and the use of the suppression pool and containment sprays for removal of aerosol and soluble gases. For the passive designs, EPRI proposed that the source term also be based on consideration of passive mitigation functions and systems such as steam condensation-

driven aerosol removal, main steam isolation valve leakage control, and secondary building fission product leakage control.

At the time that the draft Commission paper dated February 27, 1992, was published, the NRC staff was developing a revised source term based on source term calculations performed by the source term code package for individual accident sequences selected in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990. The staff concluded that the fission product release source terms proposed by EPRI and those being developed by the staff were very close for all radionuclide groups, except tellurium and low-volatile elements. The reasons for and the impact of the differences between Brookhaven National Laboratories' (BNLs') and EPRI's estimates for tellurium and the low-volatile elements were under review at the time.

The staff was in the final stage of completing its proposed update of the TID-14844 source term, including fission product removal mechanisms within the containment.

SBWR Position

A related issue is addressed in Subsection 19H.2.56. The SBWR design basis source term calculations are described in Subsection 15.6.5 and are based on EPRI-proposed source terms identified in the Utilities Requirement Document (URD). The staff's proposed update of the TID-14844 source term, documented in the June 1992 draft of NUREG-1465, is currently out for comment.

<u>Anticipated Transients Without Scram (ATWS) [I.B]</u>

Summary of Issue

As discussed in SECY-90-016, the ATWS rule (10CFR50.62) was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred.

In its June 26, 1990, Staff Requirements Memorandum (SRM), the Commission approved the staff's position that diverse scram systems should be provided for evolutionary advanced light water reactors (ALWRs), but directed that, if the applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the diverse scram system.

SBWR Position

A related issue is addressed in Subsection 19H.2.5. The SBWR design includes diverse scram systems. Analyses of ATWS events and design features for ATWS prevention and mitigation incorporated in the SBWR Standard Plant design can be found in Section 15.8.

As described in Subsection 15.8.3.7, the results of the ATWS analyses demonstrate that the proposed ATWS design for the SBWR is satisfactory in mitigating the consequences of an ATWS.

Mid-Loop Operation [I.C]

Summary of Issue

In SECY-90-016, the staff stated that it was concerned that decay heat removal capability could be lost when a pressurized water reactor (PWR) is shut down for refueling or maintenance and drained to a reduced reactor coolant system (RCS) or "mid-loop" level. The staff's position is that evolutionary pressurized water reactor (PWR) vendors propose design features to ensure high reliability of the shutdown decay heat removal system. The staff also concludes that passive plants must also have a reliable means of maintaining decay heat removal capability during all phases of shutdown activities, including refueling and maintenance.

SBWR Position

Although this issue is directed specifically to PWR vendors, the general issue regarding decay heat removal capability during reactor shutdown, refueling, or maintenance will be addressed here.

A related issue is addressed in Subsection 19H.2.17. The SBWR Standard Plant design provides for reliable decay heat removal capability during reactor shutdown by means of diversity. As noted in Subsection 9.1.3.2.3, the Fuel and Auxiliary Pools Cooling System (FAPCS) can provide backup shutdown cooling under a condition where the reactor has been depressurized and where normal shutdown cooling is not available using the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System.

Station Blackout [I.D]

Summary of Issue

As discussed in SECY-90-016, the station blackout rule (10CFR50.63) allows utilities several design alternatives to ensure that an operating plant can safely shut down in the event that all ac power (off-site and on-site) is lost. The staff concluded that the preferred method of demonstrating compliance with 10CFR50.63 is through the installation of a spare (full-capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide 1.155, and that is capable of powering at least one complete set of normal shutdown loads.

Although passive designs do not rely on active systems for safe shutdown following an event, the staff concludes that the non-safety-related diesel generators may require some regulatory oversight. This issue is enveloped for the passive designs under the issue on regulatory treatment of non-safety systems (see paragraph III.A, "Regulatory Treatment of Non-Safety Systems").

SBWR Position

A related issue is addressed in Subsection 19H.2.16. The SBWR does not require emergency ac power to achieve safe shutdown. Regulatory treatment of the SBWR non-safety-related diesel generators is addressed under item III.A.

Fire Protection [I.E]

Summary of Issue

As discussed in SECY-90-016, the staff recommended that the NRC guidance to resolve fire protection issues should be enhanced to minimize fire as a significant contributor to the likelihood of severe accidents for advanced plants.

The staff's position on this issue for passive plants is that passive plants should also be reviewed against the enhanced fire protection criteria identified in the staff's April 27, 1990, response to the ACRS's April 26, 1990, letter, which recommended staff consideration of additional matters in its evaluation of the fire protection designs.

SBWR Position

This issue is considered resolved through compliance with the following six issues identified in NUREG/CR-5088, "Fire Risk Scoping Study":

- Seismic/Fire Interactions – This issue will be evaluated as a part of probabilistic risk assessment (PRA) review and could be addressed by a walkdown.

- Fire Barrier Qualifications – This issue is addressed in the current regulations. A surveillance and maintenance program will resolve this issue.

- Manual Fire Fighting Effectiveness – This issue is addressed in the current regulations. Training for fire brigades will resolve this issue.

- Total Environmental Equipment Survival – The SBWR Standard Plant is in compliance with General Design Criterion (GDC) 3, which states that the basic design criteria and SBWR systems will be designed to meet this criteria.

- Control System Interactions – For the SBWR Standard Plant design, the independent safe shutdown capability is provided by the Remote Shutdown System (RSS).

- Improved Analytical Codes – For the SBWR Standard Plant design, the redundant safety systems are located in the separate fire areas. Hence, an improved code to show that the redundant train in the same area is protected is not needed.

Intersystem Loss-Of-Coolant-Accident [I.F]

Summary of Issue

As discussed in SECY-90-016, the staff recommended that future evolutionary ALWR designs reduce the possibility of a loss-of-coolant accident (LOCA) outside containment by designing (to the extent practicable) all systems and subsystems connected to the Reactor Coolant System (RCS) to withstand the full RCS pressure. The staff further recommended that systems that have not been designed to withstand full RCS pressure should include the following:

- The capability for leak testing of the pressure isolation valves
- Valve position indication that is available in the control room when isolation valve operators are deenergized
- High-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed

In its June 26, 1990, SRM, the Commission approved the staff's position on intersystem LOCA provided that all elements of the low-pressure system are considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets). The staff's position on this issue for passive plants is that passive plants should also be reviewed against the position for intersystem LOCA identified in the Commission's June 26, 1990, SRM.

SBWR Position

A related issue is addressed in Subsection 19H.2.44. The conclusions outlined in Subsection 19H.2.44 are considered to also resolve this issue.

Hydrogen Control [I.G]

Summary of Issue

Containments are required to be designed for control of hydrogen generation following an accident. 10CFR52.47(a)(ii) requires all applicants for design certification to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10CFR50.34(f). 10CFR50.34(f) requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction and that can ensure that uniformly distributed hydrogen concentrations in the containment do not exceed 10% (by volume) or that the post-accident atmosphere will not support hydrogen combustion.

The staff's position in SECY-90-016 is that the requirements of 10CFR50.34(f)(2)(ix) remain unchanged for evolutionary ALWRs. The staff's position on this issue for passive plants is that passive plants should also be designed, as a minimum, to perform the following tasks:

- Accommodate hydrogen equivalent to 100% metal-water reaction of the fuel cladding
- Limit containment hydrogen concentration to no greater than 10%
- Provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents

SBWR Position

Related issues are addressed in Subsections 19G.2.21 and 19H.2.19.

For a severe accident with EPRI source term assumptions, 11% metal-water reaction is considered an appropriate conservative lower bound estimate to handle the full spectrum of events in which the combustion control system may have to operate. A lower percent metal-water reaction is conservative since the SBWR containment is initially inert, and hydrogen generation will act as a diluent to the containment oxygen concentration profile.

The 10% containment hydrogen concentration limit is applicable to non-inerted containment systems, and, therefore, is not applicable to the SBWR.

The SBWR containment utilizes both inerting and hydrogen igniters for containment-wide hydrogen control for loss-of-coolant-accidents (LOCAs), as described in Subsection 6.2.5.

<u>Core Concrete Interaction – Ability to Cool Core Debris [I.H]</u>

Summary of Issue

In the unlikely event of a severe accident in which the core has melted through the reactor vessel, it is possible that containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions between the core debris and concrete can generate large quantities of additional hydrogen and other non-condensable gases, which could contribute to eventual overpressure failure of the containment. Therefore, the staff concluded that plant designs should include features to enhance core debris coolability.

As discussed in SECY-90-016, the staff recommended general criteria for evolutionary ALWR designs. The staff's position on this issue for passive plants is that passive plants should also be evaluated against these general criteria, which are as follows:

- Provide reactor cavity floor space to enhance debris spreading
- Provide a means to flood the reactor cavity to assist in the cooling process

- Protect the containment liner and other structural members with concrete if necessary

- Ensure that the containment can accommodate the pressure increases resulting from core-concrete interactions involving a range of scenarios which release core debris into the containment for 24 hours following the start of a severe accident.

SBWR Position

SBWR design features to enhance core debris coolability are assumed in the Probabilistic Risk Assessment (PRA) and are documented in Appendix 19B. These features include the following:

- Maximized reactor cavity floor space (lower drywell floor area) to improve the potential for ex-vessel debris cooling (see Subsection 19B.2.1.8)

- Lower drywell flooder system to provide automatic cavity flooding in the event of core debris discharge from the reactor vessel (see Subsection 19B.2.1.3)

- One-meter-thick sacrificial layer of concrete on the floor and walls of the lower drywell to protect the containment liner against debris attack (see Subsection 19B.2.1.7)

- The potential to manually vent the containment from the suppression chamber air space when continued core-concrete interaction (CCI) occurs (see Subsection 19B.2.2.3)

<u>High-Pressure Core Melt Ejection [I.I]</u>

Summary of Issue

In SECY-90-016, the staff recommended that evolutionary ALWR designs should include a depressurization system and cavity design features to contain ejected core debris to reduce the potential for containment failure by direct containment heating (DCH). The staff is concerned that this phenomenon might occur from the ejection of molten core debris under high pressure from the reactor vessel resulting in wide dispersal of core debris, rapid oxidation, and extremely rapid addition of energy to the containment atmosphere.

In its June 26, 1990, SRM, the Commission approved the staff's position with the directive that the cavity design, as a mitigating feature, should not unduly interfere with operations, including refueling, maintenance, or surveillance activities.

The staff's position on this issue for passive plants is that passive plants should be evaluated against the following general criteria:

- Provide a reliable depressurization system

- Provide cavity design features to decrease the amount of ejected core debris that reaches the upper containment

Depressurization of the Reactor Coolant System (RCS) is crucial to the operation of the passive safety features that limit the likelihood of core damage, as well as to reducing the potential for containment failure by DCH from the ejection of core debris at high pressure.

SBWR Position

The SBWR provides reliable RCS depressurization via the Automatic Depressurization System (ADS), as described in Subsection 6.3.3. Manual backup of the ADS is included in the SBWR Emergency Procedure Guidelines (EPGs) and is an essential action for severe accident mitigation.

As mentioned in Subsection 19B.10.2.4, the SBWR also provides the following mechanisms which may limit the transport of the molten debris from the lower cavity to the upper cavity:

- Trapping of the debris in the lower drywell inside the corium shield wall and shield plate

- Impaction, settling, and removal of the debris particles in the gas transport pathway connecting the lower and upper drywell compartments

<u>Containment Performance [I.J]</u>

Summary of Issue

As discussed in SECY-90-016, the staff recommended the use of a conditional containment failure probability (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of evolutionary ALWRs.

In its June 26, 1990, SRM, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for the evolutionary LWRs, but directed that this objective should not be imposed as a requirement and that the use of the CCFP should not discourage accident prevention. The staff was directed to review suitable alternative, deterministically established containment performance objectives providing comparable mitigation capability that may be submitted by the applicants.

The staff's position on this issue for passive plants is that passive plants should use a CCFP of 0.1 or a deterministic containment performance goal that offers comparable protection. The staff will consider any suitable alternative, deterministically established containment performance objectives providing comparable mitigation capability.

SBWR Position

The SBWR analysis of containment performance, as documented in Appendix 19B, uses a deterministic containment performance goal that offers protection comparable to a CCFP of 0.1.

Dedicated Containment Vent Penetration [I.K]

Summary of Issue

As discussed in SECY-90-016, the staff recommended the approval of the use of an overpressure protection system for the ABWR that uses a dedicated containment vent. This system is designed to avoid gross containment failure resulting from postulated slow rising overpressure scenarios that could result from postulated multiple safety system failures.

In its June 26, 1990, SRM, the Commission approved the use of the containment overpressure protection system for the ABWR, subject to a comprehensive regulatory review to weigh the "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that full capability to maintain control over the venting process is provided in the design.

At the time of issuance of the draft Commission paper (February 27, 1992), the staff had insufficient information to determine whether a containment vent is necessary for passive plant designs, and adopted the position that the need for a containment vent for the passive plant designs be evaluated on a design specific basis.

SBWR Position

A related issue is addressed in Subsection 19G.2.44. The SBWR design utilizes a manually operated valve in a Containment Atmospheric Control System (CACS) line which exhausts to the plant stack. This feature facilitates any future system enhancements which may be found necessary.

Equipment Survivability [I.L]

Summary of Issue

As discussed in SECY-90-016, the staff recommended that features provided only for severe-accident protection need not be subject to the 10CFR50.49 environmental qualification requirements, 10CFR50 Appendix B quality assurance requirements, and 10CFR50 Appendix A redundancy/diversity requirements. However, SECY-90-016 further stated that mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In instances where safety-related equipment which is provided for design bases accidents is relied upon to cope with severe-accident situations, there should also be a high confidence that this equipment will survive severe-accident conditions for the period that it is needed to perform its intended function.

In its letter dated May 6, 1991, the staff clarified its position that this criteria would be applied to those features provided only for severe-accident mitigation.

The staff's position on this issue for passive plants is that features provided only for severe-accident protection need not be subject to the 10CFR50.49 environmental qualification requirements, 10CFR50 Appendix B quality assurance requirements, and 10CFR50 Appendix A redundancy/diversity requirements.

SBWR Position

The issue of equipment survivability is the subject of current discussions between the NRC staff and GE for the ABWR design. A submittal will be made for the SBWR after this issue is resolved for the ABWR. It is expected that the scope of instrumentation and equipment required to satisfy SECY-90-016, SECY-93-087, 10CFR50.34(f) will be addressed at that time.

Elimination of Operating Basis Earthquake (OBE) [I.M]

Summary of Issue

In SECY-90-016, the staff discussed its proposal to decouple the operating basis earthquake (OBE) from the safe-shutdown earthquake (SSE) on a design-specific basis for evolutionary designs. The regulations in 10CFR100 Appendix A establish the OBE at one-half of the SSE. The staff stated that the OBE should not control the design of safety systems and was evaluating possible changes to the regulations that would reduce the magnitude of the OBE relative to the SSE.

EPRI requested that the staff evaluate the elimination of the OBE altogether from design of systems, structures, and components in nuclear power plants. The NRC staff, in evaluating the decoupling of the OBE from the SSE, is also evaluating the possibility of redefining the OBE in order to satisfy its function without an explicit response analysis.

EPRI's position on seismic design is that it is unnecessary to perform two complete sets of seismic analyses – one for the OBE and one for the SSE. The NRC staff agrees, in principle, with this position but finds that existing design practices for piping and structures do not result in designs that are significantly controlled by the OBE. As stated in SECY-90-016, certain interim measures, such as allowing higher damping values for piping analyses, have been already implemented to alleviate the situation of having the OBE significantly controlling the design.

The staff's position on this issue for passive plants is to eliminate the OBE from design of systems, structures, and components. Until the final rulemaking to 10CFR100 Appendix A is completed, the elimination of the OBE from design of passive designs will require an exemption from the current regulations with acceptable supporting justification from the designer.

SBWR Position

For the SBWR, the SSE is the earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. The SSE produces the maximum vibratory ground motion for which Seismic Category I systems and components are designed to remain functional.

The OBE is not an SBWR design requirement. Consistent with the Draft Appendix S to 10CFR50, the design requirements associated with the OBE, when the level of OBE ground motion is chosen to be one-third of the SSE ground motion, are satisfied without performing explicit response or design analyses.

Further details regarding SBWR seismic design can be found in Section 3.7.

Inservice Testing of Pumps and Valves [I.N]

Summary of Issue

As discussed in SECY-90-016, the staff recommended that the following provisions be applied to all safety-related pumps and valves and not limited to ASME Code Class 1, 2, and 3 components.

- Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps and check valves.

- Designs should incorporate provisions to test motor operated valves under design basis differential pressure.

- Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques.

The staff concluded that these requirements were necessary to provide an adequate level of assurance of operability.

In its June 26, 1990, SRM, the Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

The staff's position on this issue for passive plants is that the above requirements also be imposed on passive ALWRs, since the passive safety systems rely on the proper operation of this equipment (i.e., check valves, depressurization valves) to mitigate the effects of transients.

SBWR Position

A detailed description of inservice testing of pumps and valves for the SBWR Standard Plant is contained in Subsection 3.9.6.

The SBWR safety-related pumps and piping configurations accommodate inservice testing at a flowrate at least as large as the maximum design flow for the pump.

All SBWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design conditions. Inservice testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves.

The motor operated valve (MOV) equipment specifications require the incorporation of the results of either in-situ or prototype testing with full flow and pressure or full differential pressure to verify the proper sizing and correct switch settings of the valves.

The establishment of a program to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques is the responsibility of the combined operating license (COL) applicant. Refer to Subsection 3.9.7.3 for COL license information related to this issue.

Industry Codes and Standards [II.A]

Summary of Issue

In SECY-91-273, the staff raised the concern that a number of design codes and industry standards dealing with new plant construction have been recently developed or modified, and that the NRC has not yet determined their acceptability. EPRI and ALWR vendors are using codes and standards in their applications that the staff has not endorsed.

The staff's position is that the newest codes and standards that have been endorsed by the NRC be used in reviews of both evolutionary and passive plant design applications. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis.

SBWR Position

Industry codes and standards applicable to the design of the SBWR Standard Plant are listed in Table 1.9-3, and are submitted for NRC approval.

Electrical Distribution [II.B]

Summary of Issue

In SECY-91-078, the staff recommended that evolutionary plant designs should include the following:

- An alternate power source to the non-safety loads, unless the design can demonstrate that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs

- At least one off-site circuit to each redundant safety division supplied directly from one of the off-site power sources with no intervening non-safety buses in such a manner that the off-site source can power the safety buses upon a failure of any non-safety bus

Because the passive designs do not rely on active systems for safe shutdown following an event, the staff, at the time of issuance of the draft Commission paper (February 27, 1992), had not determined the applicability of this issue to the passive designs. This issue is enveloped for the passive designs under the issue on regulatory treatment of non-safety systems (see paragraph III.A).

SBWR Position

The SBWR Standard Plant design does not rely on active systems for safe shutdown following an event. Regulatory treatment of the SBWR non-safety systems is addressed in paragraph III.A.

Seismic Hazard Curves and Design Parameters [II.C]

Summary of Issue

To assess the seismic risk associated with an ALWR design, EPRI has proposed the use of generic bounding seismic hazard curves for sites in the central and eastern United States. EPRI proposes that these curves be used in the seismic probabilistic risk assessment (PRA). The regulations do not require, and at the time of issuance of the draft Commission paper (February 27, 1992), the staff did not intend to require, that a seismic PRA be performed to determine if a site is acceptable.

Based on the staff's review of historical seismicity and Lawrence Livermore National Laboratories (LLNL) hazard estimates, the staff concluded that the EPRI seismic hazard bounding curve is not sufficiently conservative. At the time of issuance of the draft Commission paper (February 27, 1992), the staff was evaluating the seismicity and ground motion inputs used in the LLNL and EPRI studies to determine if the uncertainties in the curves could be reduced.

As part of the COL process, the applicant will have to demonstrate that the site-specific seismic parameters meet the certified design parameters to ensure issue preclusion at the COL hearing. Should an actual site value exceed the design envelope in a certain area, a specific analysis will have to be performed to verify that the design is still acceptable for that site.

SBWR Position

Generic bounding seismic hazard curves proposed by EPRI for use in the seismic PRA are no longer included in the current version of the Utilities Requirement Document (URD). Assessment of seismic risk for the SBWR Standard Plant design is performed using seismic margin analysis, rather than seismic PRA. Seismic margin analysis does not require the use of the EPRI generic bounding seismic hazard curves.

See Subsection 2.7.4 for COL license information related to this issue.

<u>Leak-Before-Break [II.D]</u>

Summary of Issue

Under the broad scope revision of General Design Criterion (GDC) 4 (52FR41288, October 27, 1987), the NRC allows the use of advanced technology to exclude from structural design consideration the dynamic effects of pipe ruptures in nuclear power plants provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. Demonstration of low probability pipe rupture requires a deterministic fracture mechanics analysis that evaluates the stability of postulated small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaw could grow to unstable sizes. The concept underlying such analyses is referred to as "leak-before-break (LBB)."

At the time of issuance of the draft Commission paper (February 27, 1992), the LBB approach had been approved by the NRC staff for then currently operating and near-term operating licensed nuclear power plants based on a case-by-case review of plant-specific analyses. As discussed in SECY-89-013, the staff will evaluate the acceptability of the use of LBB considerations in the ALWR designs when it can be justified.

The staff concluded that the limitations and acceptance criteria for LBB applications in ALWRs are the same as those established for currently operating nuclear power plants. The staff approves the application of the LBB approach to both evolutionary and passive ALWRs seeking design certification under 10CFR52 when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the combined license phase by performing the appropriate inspections, tests, analyses, and acceptance criteria (ITAAC).

SBWR Position

Subsection 3.6.3 and Appendix 3C describe the implementation of the LBB evaluation procedures for SBWR as permitted by the broad scope amendment to GDC-4. An LBB report shall be prepared by the COL applicant, along with the stress report for the LBB-qualifiable piping in accordance with the guidelines presented in Appendix 3C. The LBB-qualified piping will be excluded from pipe breaks for design against their potential dynamic effects.

Subsection 3.6.3 describes (1) certain design bases where the LBB approach is not recognized by the NRC as applicable for exclusion of pipe breaks and (2) certain conditions which limit the LBB applicability. Appendix 3C provides guidelines for LBB applications describing in detail the following necessary elements of an LBB report to be submitted by the COL applicant for NRC approval.

Classification of Main Steamlines of Boiling Water Reactors (BWRs) [II.E]

Summary of Issue

Because of recurring problems with excessive leakage of main steam isolation valves (MSIVs) in BWR plant designs, Regulatory Guide 1.´ "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," recommended the installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified limits. Operating experience with the LCS has required substantial maintenance and worker exposure. Additionally, the NRC has generic concerns with the effectiveness of the LCS to perform its intended function under conditions of high-MSIV leakage.

These concerns led EPRI to propose an alternative approach to ensure that doses associated with MSIV leakage would be acceptably low. The resolution proposed by EPRI eliminates the safety-related LCS, allows higher leakage limits through the MSIVs, and uses an alternate MSIV leakage treatment method.

Section 3.2.2 of the Standard Review Plan (SRP) recommends that the main steamline from the outermost isolation valve up to, but not including, the turbine stop valve including branch lines up to the first valve, be classified as Quality Group B (Safety Class 2). Regulatory Guide 1.29 designates such piping as Seismic Category I. The staff concludes that the main steam piping from the outermost isolation valve up to the seismic interface restraint and branch lines up to the first closed valve should conform to Appendix A of Section 3.2.2 of the SRP and Regulatory Guide 1.29. The main steamline from the seismic interface restraint up to but not including the turbine stop valve should be classified as Quality Group B but may be classified as non-seismic Category I. However, all pertinent quality assurance requirements of Appendix B to 10CFR50 are applicable to this portion of the main steamline from the seismic interface restraint to the turbine stop valve. These requirements are needed to ensure that the quality of the piping material is commensurate with its importance to safety during both operational and accident conditions.

To ensure the integrity of the bypass piping from the first valve to the main condenser hotwell, the staff and EPRI both agree that preventing gross structural failure of the piping and hotwell would provide assurance that leakage from the MSIVs following a design basis accident would not exceed the 10CFR100 guideline. The issue remaining is the classification of the main steam bypass piping between the first normally closed valve and the condenser hotwell as well as the hotwell itself. The staff proposes that the main steam bypass line from the first valve up to the condenser inlet and the piping between the turbine stop valve and the turbine inlet should not be classified as safety related nor as seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate its structural integrity under SSE loading conditions.

The staff proposes that the condenser be seismically analyzed to ensure that it is capable of maintaining its structural integrity during and after the SSE. Since the dose analysis considers that the condenser is open to atmosphere, it is only necessary to ensure there is no gross structural failure of the condenser.

Overall, the staff concludes that the above described approach for both evolutionary and passive ALWRs to resolve the BWR main steamline classification issue provides reasonable assurance that the main steam piping from the outermost isolation valve up to the turbine stop valve, the main steam bypass line up to the condenser, and the main condenser will retain their pressure and structural integrity during and following a safe shutdown earthquake.

SBWR Position

The SBWR design does not include an MSIV LCS, as noted in Section 6.7, and adopts the EPRI alternative approach.

According to Table 3.2-1, the Turbine Bypass System lines and the branch line of the main steamline, including supports between the second isolation valve and the turbine stop valve from the branch point at the main steamline to, and including, the first valve in the branch line, are categorized as Quality Group B, non-safety, and non-seismic. Non-seismic structures and equipment are those which do not fall into seismic Category I or II definitions. The main steamlines from the containment outboard isolation valves and all branch lines 2-1/2 inches in diameter and larger, up to and including the first valve (including lines and valve supports) are designed by the use of an appropriate dynamic seismic system analysis to withstand the SSE design loads in combination with other appropriate loads, within the limits specified for Class 2 pipe in the ASME Code, Section III.

The main condenser is classified as non-safety related and non-seismic Category I. However, the supports and anchors for the main condenser are designed to withstand an SSE. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) are provided for the seismic capability design commitment of the supports and anchors in the SBWR Tier 1 Design Certification Document.

Tornado Design Basis [II.F]

Summary of Issue

At the time of issuance of the draft Commission paper (February 27, 1992), the NRC regulatory position with regard to design basis tornadoes is contained in two documents issued in 1974, WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," and Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." Regulatory Guide 1.76, in particular, delineates the maximum wind speeds of 240 to 360 mph depending on the regions.

After reviewing updated tornado data and the analysis provided in NUREG/CR-4661, "Tornado Climatology of the Contiguous United States," dated May 1986, the staff concluded that it is acceptable to reduce the tornado design basis wind speeds to 200 mph for the United States west of the Rocky Mountains, and to 300 mph for the United States east of the Rocky Mountains.

The staff's position is that a maximum tornado wind speed of 300 mph be used for the design basis tornado to be used in the design of evolutionary and passive ALWR designs. As part of the COL process, the applicant will have to demonstrate that a design capable of withstanding a 300 mph tornado will also be sufficient to withstand other site hazards. Should an actual site hazard exceed the design envelope in a certain area, a specific analysis will have to be performed to verify that the design is still acceptable for that site.

SBWR Position

As noted in Subsection 2.3.1, the design basis tornado maximum wind speed is 300 mph (134 m/s), consistent with the staff's position. Subsection 3.3.3 contains COL license information related to this issue.

Containment Bypass [II.G]

Summary of Issue

The phenomenon of containment bypass is associated with the failure of the containment system to channel fission product releases through the suppression pool, or the failure of passive containment cooling system heat exchanger tubes in the large pools of water outside the containment. Leakage paths could exist in the pathway between the drywell and the wetwell airspace that could allow steam to bypass the suppression pool and might overpressure the containment. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell structure, and penetrations through the drywell structure. In addition, a containment design which uses an external heat exchanger carries with it a potential of containment bypass from a leak in the heat exchanger. High temperatures associated with severe accidents or core debris carried from the reactor vessel could threaten the integrity of the heat exchanger tubes, and therefore provide a pathway for the release of fission products. Bypass of internal suppression pools could lead to overpressurization of the containment, and threaten its integrity. The staff believes that vendors should make reasonable efforts to minimize the possibility of bypass leakage, and should also allow for a certain amount of leakage in the containment design.

The provision of containment sprays in the drywell and/or wetwell would also reduce the impact of suppression pool bypass leakage on containment performance. In view of the contribution they can make to accident management, the staff, at the time of issuance of the draft Commission paper (February 27, 1992), was evaluating the need for containment spray systems for all ALWRs.

SBWR Position

The SBWR vacuum relief valves are high-reliability, leak-proof components, and do not provide a likely potential for steam bypass. The containment vent wall and diaphragm slab are of leak-tight construction, and also do not provide a likely potential for steam bypass. The leakage potential of the liner plate and penetrations is evaluated in Subsections 19B.3.3.1 and 19B.3.3.2, respectively. The results of the analyses (Subsection

19B.3.3.3) show that no liner leakage will occur before the capability pressure is reached, and leakage through fixed (mechanical and electrical) penetrations is negligible.

Isolation condenser/pressure containment cooling (IC/PCC) tube failure due to fission product plugging is evaluated in Appendix 19BC. As noted in Subsection 19BC.7, the IC/PCC tubes will not fail because of fission product plugging during a severe accident. Test data from the Pre-Service Inspection (PSI) test program will be used to make more definitive judgments about possibility of IC/PCC tube plugging and failure. In addition, the SBWR long-term cooling model of the containment accounts for postulated leakage between the drywell and suppression chamber gas space (see Subsection 6.2.1.1.3.2).

As noted in Subsection 6.5.2, the SBWR contains both suppression chamber and upper drywell containment sprays. Neither spray system is safety-related, and no credit is taken for fission product removal under design basis accident evaluations. However, in a severe accident scenario, operation of the Fuel and Auxiliary Pool Cooling System (FAPCS) in the drywell spray mode may be required to maintain containment integrity. The drywell spray mode reduces the consequences of suppression pool bypass. Manual initiation of the drywell sprays upon an indication of increasing drywell temperatures is an appropriate and essential mitigation strategy. This action is contained in the SBWR Emergency Procedure Guidelines (EPGs).

Containment Leak Rate Testing [II.H]

Summary of Issue

EPRI proposed that the maximum interval between Type C leakage rate tests should be 30 months rather than the 24-month maximum interval required, at the time of issuance of the draft Commission paper (February 27, 1992), in Appendix J to 10CFR50 for both evolutionary and passive plant designs. This proposal was generated to allow some margin between the nominal 24-month refueling interval and the Type C test interval to ensure that plant shutdowns will not be required solely to perform Type C tests.

At the time of issuance of the draft Commission paper (February 27, 1992), the staff had developed proposed changes to Appendix J of 10CFR50 for all reactors, and were sent to the Commission in SECY-91-348.

The staff recommends that, until the rule change proceedings for Appendix J of 10CFR50 are completed, the maximum interval between Type C leakage rate tests for both evolutionary and passive plant designs be 30 months rather than the 24-month maximum interval.

SBWR Position

A related issue is addressed in Subsection 1A.2.34. As noted in Subsection 1A.2.34, the maximum interval between Type C leakage rate tests is 30 months, consistent with the staff's position.

Post-Accident Sampling System [II.I]

Summary of Issue

Regulatory Guide 1.97 and NUREG-0737, "Clarification of TMI Action Plan Requirements," provide guidance regarding the design of the post-accident sampling system (PASS) used to implement regulation 10CFR50.34(f)(2)(viii).

EPRI has proposed deviation from the design requirements for the PASS in the following areas:

- Elimination of the Hydrogen Analysis of Containment Atmosphere Samples

  EPRI has stated that the hydrogen analysis of the containment atmosphere can be accomplished by the safety-grade containment hydrogen monitor required by 10CFR50.34(f)(2)(xvii) and II.F.1 of NUREG-0737. The staff concludes that the safety-grade instrumentation provides acceptable justification for requesting this deviation, and does not consider this request to be a policy matter.

- Elimination of Dissolved Gas and Chloride Analyses of Reactor Coolant Samples

  EPRI considers the analyses of the reactor coolant for dissolved gas and chloride to be unnecessary because gases accumulated in the reactor vessel (mainly hydrogen) will be removed by venting, and corrosion due to the presence of chloride and oxygen will be minimized by prompt depressurization and cooling. Additionally, the amount of dissolved hydrogen in the reactor coolant can be determined based upon the hydrogen concentration measured in the containment atmosphere. 10CFR50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737 specify that the PASS should have the capability to analyze dissolved hydrogen, oxygen, and chloride.

  The staff concludes that even with vented reactor vessels, there are some postulated accident sequences in which the reactor coolant system is intact at reduced pressure, and heat is being removed. For these cases, it will not be possible to evaluate concentrations of the dissolved gases in the reactor coolant from their concentrations measured in the containment. Therefore, the staff's position is that the requirement for PASS sampling of coolant should not be eliminated. However, the staff agrees that sampling 24 hours after the end of power operation would be adequate to ensure long-term decay heat removal.

- Relaxation in the Time Requirement for Sampling Activity Measurements

  EPRI states that if boron solution has been added to permit plant shutdown, reactor water samples can be taken for boron analyses starting 8 hours after the end of power operation. EPRI states that the samples for activity measurements will not be required for 24 hours after the accident.

  Item II.B.3 of NUREG-0737 specifies that the PASS should have the capability to obtain coolant and containment atmosphere sampling results within 3 hours from the time after the accident.

Based on commitments and justifications from EPRI, the staff concurs with EPRI that boron sampling will not be required for the first 8 hours after an accident. The staff also concludes that the requested extension of time for sampling activity measurements 24 hours after an accident is acceptable. The staff's position is to allow for deviation from the requirements of Item II.B.3 of NUREG-0737.

SBWR Position

- Elimination of the Hydrogen Analysis of Containment Atmosphere Samples

  Analysis of hydrogen in SBWR containment atmosphere samples is provided by the Containment Atmosphere Monitoring System (CAMS).

- Elimination of Dissolved Gas and Chloride Analyses of Reactor Coolant Samples

  During a core uncovering accident, the accumulation of noncondensible gases in the SBWR reactor vessel will be prevented. Also, excessive corrosion in the SBWR reactor vessel will not occur because corrosive conditions will be prevented and the SBWR reactor vessel will not be kept in a hot, pressurized condition.

  During the early hours of a design basis loss-of-coolant-accident (LOCA), any hypothesized accumulation of gases in the SBWR reactor vessel would be prevented by the opening of the six Automatic Depressurization Subsystem (ADS) valves that vent the steam produced by core decay heat out of the reactor vessel and into the containment (see SBWR SSAR, Subsection 6.3.3). Reactor vessel depressurization then allows nitrogen-saturated demineralized water to flow into the reactor vessel from the Gravity-Driven Cooling System (GDCS). Noncondensible gases will be contained in this stream and will also be formed from radiolysis reactions. Any noncondensible gases will be rapidly stripped from the reactor vessel water as boiling occurs and will be swept into the containment by the escaping steam.

  Corrosion of the SBWR reactor vessel and its components will be held to negligible amounts by the degassification processes discussed above and by prompt depressurization and cool-down. Thus, the capability to analyze reactor water for dissolved hydrogen, dissolved oxygen, dissolved total gases, and chloride are neither needed for mitigation of the accident nor provided by the SBWR PASS.

- Relaxation in the Time Requirement for Sampling Activity Measurements

  During an accident, the immediate responses of the plant operators are discussed in the SBWR SSAR, Subsection 18.4.2.11, Safety Parameter Display System, and Subsection 18A.2, RPV Control Guideline. Later, to assist in planning the accident recovery program, sampling of reactor coolant for gross activity and isotropic analyses will be performed. These data will not be needed by the plant operators during the initial phase of an accident.

Sampling and analysis for reactor coolant boron concentration may be required in some cases. Provision is made to perform this sampling and analysis at 8 hours or later after the core uncovering occurred.

Level of Detail [II.J]

Summary of Issue

In its February 15, 1991, SRM on SECY-90-377, the Commission provided guidance regarding the level of detail of information required to determine the adequacy of design certification applications under 10CFR52. Although the level of detail issue is applicable to all design certification applications, the staff has been reviewing the ABWR as the lead plant in resolving this issue.

In a meeting with GE, senior NRC managers and the vendor discussed certain areas of review for which the designer has not provided final design details. The staff and GE agreed to pursue the development of design acceptance criteria (DAC) with associated NRC "check points" as a substitute for detailed design information for a few limited areas of the design. These issues would be documented in the Safety Analysis Report and the inspections, tests, analyses, and acceptance criteria (ITAAC), as appropriate.

The staff concludes that the level of detail issue is applicable to all design certification applications, but expects it to be resolved in the context of the ABWR review.

SBWR Position

The level of detail of information contained in GE's SBWR certification documents (e.g., SSAR and Tier 1 Design Certification Document) is consistent with the level of detail contained in GE's ABWR certification documents.

Prototyping [II.K]

Summary of Issue

SECY-91-074 discussed the process that the staff will use for determining the need for a prototype or other demonstration facility for the advanced reactor designs. The staff stated it will follow the procedure outlined in the paper to determine the various types of testing, up to and including a prototype facility, that may be needed to demonstrate that the advanced reactor designs are sufficiently mature to be certified.

SECY-91-273 stated that the necessity for separate effects and scaled integral testing for passive designs was under consideration. At the time of issuance of the draft Commission paper (February 27, 1992), this issue remained under consideration, as it applied to passive designs.

SBWR Position

The overall SBWR program includes various separate effects and scaled integral test programs to demonstrate that the SBWR is sufficiently mature to be certified.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) [II.L]

Summary of Issue

SECY-91-178 provided the staff's recommendations concerning the form and content of the ITAAC for a design certification rule and combined license as required by 10CFR52. In its September 24, 1991, SRM, the Commission provided guidance regarding development of ITAACs for final design approval/design certification applications.

At the time of issuance of the draft Commission paper (February 27, 1992), GE had been identified by the Nuclear Management and Resources Council (NUMARC) as the industry lead for developing ITAAC submittals.

The staff concludes that the ITAAC issue is applicable to all final design approval/design certification applications, but expects it to be resolved in the context of the ABWR review.

SBWR Position

The form and content of GE's SBWR ITAACs (included in the SBWR Tier 1 Design Certification Document) is consistent with the form and content of GE's ABWR ITAACs.

Reliability Assurance Program (RAP) [II.M]

Summary of Issue

In SECY-89-013, the staff stated that a program to ensure that the design reliability of safety significant systems, structures, and components is maintained over the life of a plant, referred to as the reliability assurance program (RAP), would be required for design certification.

The staff is working on the development of a detailed guidance document for the development of a RAP for ALWRs. The staff views the RAP for ALWRs as a program that exists at two distinct levels: The first level applies to vendor submittals for final design approval/design certification; the second level is applicable to a referencing applicant for a construction and operating license. The first level involves a top-level program that defines the scope, conceptual framework, and essential elements of an effective RAP. The second level fully develops and implements the program based on the plant-specific design information.

At the time of issuance of the draft Commission paper (February 27, 1992), the staff was working with EPRI and the ALWR vendors on the development of the first level RAP for ALWRs.

SBWR Position

Reliability Assurance Programs are provided in the SBWR Standard Plant design. The SBWR Design Reliability Assurance Program (D-RAP), described in Section 17.3, is the top-level (first level) RAP for the SBWR. The second-level owner-implemented RAP for the SBWR is referred to as the Operational Reliability Assurance Program (O-RAP), and is described in Subsection 17.3.10.

Site-Specific Probabilistic Risk Assessments [II.N]

Summary of Issue

10CFR52.47 requires all applicants seeking standard design certification to provide a probabilistic risk assessment (PRA). However, details of the specific site characteristics where a plant would be sited are not required until the combined operating license (COL) licensing stage. The staff's position is that site-specific PRA information be submitted at the COL stage that addresses applicable site-specific PRA information such as river flooding, storm surge, tsunami, vulcanism, and hurricanes, and that enveloping analyses for seismic events and tornadoes be required from the final design approval/design certification applicants.

SBWR Position

The SBWR PRA results are calculated for an average or typical site, as outlined in Appendix 19E. Although these results form a good basis for assessing the general SBWR capability to satisfy off-site dose-related goals, they do not form a basis for concluding that the SBWR would meet dose-related goals at a specific site whose characteristics cannot be defined at the point of SBWR certification.

Section 2.7 specifies COL license information related to this issue.

Severe Accident Mitigation Design Alternatives (SAMDAs) [II.O]

Summary of Issue

The staff, after *Limerick Ecology Action v. NRC*, 869 F.2d 719 (3rd Cir. 1989), concluded that a National Environmental Policy Act (NEPA) evaluation in the form of an environmental impact statement that considered severe accident mitigation design alternatives (SAMDAs) would be an essential element of an application for a combined license under Subpart C of 10CFR52, for those applications that reference a design certified under Subpart B.

In SECY-91-229, the staff requested the ALWR vendors to assess SAMDAs for their designs and provide their rationale for determining whether the SAMDAs would improve the safety of those designs.

The staff concludes that the SAMDAs issue is applicable to all final design approval/design certification applications, but expects it to be resolved in the context of the ABWR and System 80+ reviews.

SBWR Position

The assessment of SAMDAs for GE's SBWR is consistent with the assessment of SAMDAs contained in GE's ABWR certification documents.

Generic Rulemaking Related to Design Certification [II.P]

Summary of Issue

SECY-91-262 provided the Commission with the staff's recommendations regarding generic rulemaking related to design certification. As discussed in SECY-91-262, the staff concludes that consideration of generic rulemaking in lieu of design-specific rulemaking is applicable to all final design approval/design certification applications. However, the design of the passive plants was not sufficiently developed at the time of issuance of the draft Commission paper (February 27, 1992) for the staff to determine whether generic rulemaking should be initiated for the passive plant designs. An example of a generic rulemaking activity is the evaluation of source terms during postulated severe accidents.

SBWR Position

The SBWR Standard Plant design complies, in general, with NRC generic rulemaking policies, as well as design-specific rulemaking policies, required for design certification that is considered applicable to the SBWR.

Regulatory Treatment of Non-Safety Systems [III.A]

Summary of Issue

Associated with the new, passive design approach, the licensing design basis analysis relies solely on the passive safety systems to demonstrate compliance with the acceptance criteria for various design basis transients and accidents. However, uncertainties remain concerning the performance of the unique passive features and overall performance of core and containment heat removal because of lack of a proven operational performance history. The staff's review of the passive designs requires a review of not only the passive safety systems, but also the functional capability and availability of the active non-safety systems to provide significant defense-in-depth and accident and core damage prevention capability.

In addition, the staff was evaluating, at the time of issuance of the draft Commission paper (February 27, 1992), the need to establish reliability-based technical specifications for passive designs to determine which systems and components (including certain non-safety systems) will require the imposition of technical specifications, and the parameters of the technical specifications (length, surveillance, etc.). The Reliability Assurance Program is expected to strongly influence the technical specifications.

Since the passive ALWR design philosophy departs from licensing practices for evolutionary designs, new regulatory and review guidance is necessary so that the staff can appropriately review the passive vendor submittals. At the time of issuance of the draft Commission paper (February 27, 1992), significant decisions needed to be made concerning the scope of staff review of the non-safety systems and reliance on the passive safety systems.

At the time of issuance of the draft Commission paper (February 27, 1992), the staff was still evaluating this issue for the passive plant designs.

SBWR Position

Adequate capability and availability of the passive safety and active non-safety systems (when called upon) in the SBWR Standard Plant design are ensured through the establishment of functional performance requirements and acceptance criteria.

Any forthcoming regulatory requirements regarding the capability and availability of the SBWR passive safety and active non-safety systems will be reviewed by GE, and compliance to these requirements will be assessed and submitted for NRC review.

Definition of Passive Failure [III.B]

Summary of Issue

A passive failure in a fluid system means a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. In licensing reviews prior to the time of issuance of the draft Commission paper (February 27, 1992), the staff had been inconsistent in its treatment of passive failures in fluid systems in that in certain cases it imposed a passive failure in addition to the initiating event while in others it did not. The staff had determined that in most instances the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the single failure criterion to ensure the safety of a nuclear power plan..

In particular, staff pracuce had normally been to treat check valves, except for containment isolation systems, as passive devices rather than active devices during transients or design basis accidents. However, the staff, at the time of issuance of the draft Commission paper (February 27, 1992), was considering redefining failure of check valves to that of an active failure. This would cause these valves to be evaluated in a more stringent manner than that of previous licensing reviews.

At the time of issuance of the draft Commission paper (February 27, 1992), the staff was still evaluating this issue for the passive plant designs.

SBWR Position

The SBWR treatment of single component failures in fluid systems is addressed in Subsection 3.6.1. Single component failures are assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure.

Thermal-Hydraulic Stability of the SBWR [III.C]

Summary of Issue

At the time of issuance of the draft Commission paper (February 27, 1992), the analytical codes that GE used to demonstrate stability of the SBWR design had not been adequately validated. As discussed in SECY-91-273, the staff had determined that an early NRC assessment was needed of the vendor's analytical and experimental basis for demonstrating nuclear/thermal-hydraulic stability, and to identify any tests or analyses that may be needed to support staff technical evaluations of the issue.

GE had identified existing experimental data which they believe to be appropriate validation of codes to be used for stability studies. However, at the time of issuance of the draft Commission paper (February 27, 1992), GE had not provided sufficient information to permit NRC evaluation of the applicability and sufficiency of the experiments they had identified for use during code validation. Until these experiments are reviewed by the NRC, the potential need for additional experiments to support stability evaluations for design certification remains open.

SBWR Position

In Section 4D.3 of the SBWR SSAR, GE presented the test data from the Hitachi's small-scale natural circulation test loop in Japan, which GE then believed to be appropriate validation of the analytical code to be used for SBWR geysering instability studies. In this test, Freon was used as a coolant. Based on further review, GE decided to use water data instead of the Freon data for code qualification. One of the reasons was that conversion of the Freon data to the water-equivalent results introduces greater uncertainty for qualification of the analytical code.

Aritomi (J. H. Chiang, M. Aritomi, R. Inoue and M. Mori, "Thermo-Hydraulics during Startup in Natural Circulation Boiling Water Reactors," NURETH-5,.9/92) has performed the small-scale experiments using water as a coolant which show the geysering-type phenomenon. GE has performed TRACG analysis of some of the test data to demonstrate the capability of the code to predict this behavior. Results of the analysis were reported in Section 5.6, "TRACG Qualification Licensing Topical Report," NEDE-32177P, February 1993. From this analysis, it was concluded that TRACG successfully calculated the geysering oscillations seen in the experiment.

For additional information related to the SBWR stability issue, refer to the GE responses for the NRC RAIs SRXB.17 and SRXB.18.

Safe Shutdown Requirements [III.D]

Summary of Issue

General Design Criterion (GDC) 34 requires that a residual heat removal (RHR) system be provided to remove residual heat from the reactor core so that specified acceptable fuel design limits (SAFDLs) are not exceeded. Regulatory Guide (RG) 1.139 and Branch Technical Position (BTP) 5-1 implement this requirement and set forth conditions to cold shutdown (200°F for a PWR and 212°F for a BWR) using only safety grade systems within 36 hours.

Passive designs use passive heat removal systems for decay heat removal. They are restricted by the inherent ability of the passive heat removal processes and cannot reduce the temperature of the reactor coolant system below the boiling point of water for the heat to be transferred to the in-containment refueling water storage tank (IRWST) of the AP600 or the isolation condenser of the SBWR. Even though active shutdown cooling systems are available to bring the reactor down to cold shutdown or refueling conditions, these active RHR systems are not safety grade, and do not comply with the guidance of RG 1.139 or BTP 5-1.

EPRI had stated that it is not necessary for the passive safety systems to be capable of achieving cold shutdown because they believed that the passive decay heat removal (DHR) systems have an inherent high long-term reliability. In addition, the EPRI Requirements Document defines safe shutdown as 400°F.

The staff, at the time of issuance of the draft Commission paper (February 27, 1992), was evaluating the EPRI position with respect to this issue to assess the acceptability of their proposed alternative approach for meeting GDC 34. There were several issues that the staff identified as "musts" to resolve before reaching a final position on this matter, including reliability criteria for the non-safety systems which have the capability to bring the plant to cold shutdown and the acceptability of 400°F as a safe long-term state.

SBWR Position

The SBWR Standard Plant design utilizes features which are consistent with the NRC's safe shutdown requirements. The SBWR Isolation Condenser System (ICS) is designed and qualified as a safety-related system to comply with GDC 34 (see Subsection 5.4.6.3). The SBWR Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System is designed to bring the plant to cold shutdown in 36 hours in conjunction with the ICS, with loss of preferred off-site ac power, and assuming the most restrictive single active failure (see Subsection 5.4.8.2).

<u>Control Room Habitability [III.E]</u>

Summary of Issue

GDC 19 states that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. In Section 6.4 of the SRP, the staff defines this dose criterion in terms of specific whole body and organ doses (5 rem to whole body and 30 rem each to thyroid and skin).

EPRI had proposed the exposure limit for control room operators to be 5 rem whole body, 75 rem beta skin dose, and 300 rem thyroid dose.

The staff concluded that the thyroid dose limit of 300 rem and the skin dose of 75 rem (with no protective clothing) specified in the Requirements Document without further technical justifications does not meet the regulatory requirements and, therefore, is not acceptable.

The staff's position is as follows:

- That the analyses of control room habitability be based on the dose criterion defined in GDC 19 of Appendix A to 10CFR50 and Section 6.4 of the SRP (5 rem to whole body, and 30 rem each to thyroid and skin)

- That the analyses of control room habitability should be based on the duration of the accident in accordance with GDC 19 of Appendix A to 10CFR50

SBWR Position

The analyses of SBWR control room habitability are based on the dose criterion defined in GDC 19 of Appendix A to 10CFR50 and Section 6.4 of the SRP, and based on the duration of the accident in accordance with GDC 19 of Appendix A to 10CFR50, consistent with the staff's position.

Radionuclide Attenuation [III.F]

Summary of Issue

EPRI and the passive ALWR designers rely on assumptions involving fission product removal processes inside containment by natural removal effects and holdup by the secondary building and piping systems. A containment spray system is not included in the EPRI Requirements Document for passive plant designs. The staff was concerned that use of the auxiliary building for holdup may require additional restrictions be placed on the auxiliary building during normal operation that the licensee may have difficulty complying with.

In addition to evaluating the need for a containment spray system for the passive plant designs, the staff was also evaluating whether credit for the fission product attenuation in the main steamlines and condenser is appropriate for the passive BWR design because the main steamlines downstream of the main steam isolation valves and associated condenser are not designed to withstand the safe shutdown earthquake (SSE) as defined in Section III.c of 10CFR100.

SBWR Position

Use of the auxiliary building for fission product holdup is an issue that is generally applicable to PWRs. For the SBWR, the safety envelope contains, dilutes, and holds up any leakage from the containment, and has restrictions and testing criteria applied, as described in Subsection 6.2.3.

The issue regarding fission product attenuation in the main steamlines is addressed by GE for the ABWR Standard Plant design. The SBWR position is consistent with the ABWR position on this issue.

Simplification of Off-Site Emergency Planning [III.G]

Summary of Issue

EPRI had proposed to significantly simplify off-site emergency planning for passive designs because of EPRI's estimated low probability of core damage and, in the event of a core damage accident, the assurance of containment integrity and low off-site dose.

During a January 30, 1992, meeting with the staff, EPRI proposed to work with the staff to define a process for addressing simplification of emergency planning. The results of this effort was to be used as input to a generic rulemaking proposal to be initiated by NUMARC.

The staff concluded that certain modifications from the emergency planning requirements of 10CFR50 and from the siting criteria in 10CFR100 may be appropriate for the passive designs based on their unique characteristics. However, an agency determination on these issues would require evaluation of detailed design information. The staff concluded that the unique characteristics of these designs should be taken into account in determining the extent of emergency planning requirements in the plume exposure pathway emergency planning zone. A plant's ability to prevent the significant release of radioactive material or to provide very long delay times prior to a release for all but the most unlikely events should be reflected in any decision on emergency planning requirements for the passive design. However, the staff requires a high degree of assurance that all potential containment bypass accident sequences have a very low likelihood before relaxing emergency planning requirements. This issue is also complicated by the fact that the promulgation of emergency planning requirements following the TMI-2 accident was not premised on any specific assumptions about severe accident probability. Hence, it may be, as a policy matter, that even very low calculated probability values should not be a sufficient basis for changes to emergency planning requirements.

The staff had planned to evaluate this issue for the passive plant designs when sufficient supporting information was available.

SBWR Position

Off-site emergency planning is not within the scope of the SBWR Standard Plant design, as noted in Section 13.3.

RAI Number: EELB.3

Question:

If not addressed in the forthcoming SSAR Section 1.8, "Interfaces for Standard Design," and Section 1.9, "Conformance with Standard Review Plan," then please provide an explanation and how the SBWR incorporates into the design operational experience (see staff requirements memoranda (SRM) dated February 15 and March 5, 1992).

GE Response:

An explanation of how the SBWR incorporates into its design operational experience to address the Commission's concerns identified in SRMs dated February 15 and March 5, 1991, is provided in Subsection 1.9.3 of the SBWR SSAR, February 28, 1993 submittal.

RAI Number: EMCB.1

Question:

SSAR Section 4.5.1, Control Rod System Structural Materials

SSAR Paragraph 4.5.1.1, "Materials Specifications," states that the following cobalt based materials will be used in the CRD system:

- guide Roller - Stellite No. 3;
- guide Roller Pin - Haynes Alloy No. 25;
- guide Shaft - Stellite No. 6; and
- guide Shaft Bushing - Stellite No. 12.

The use of cobalt should be avoided except in cases where no other alternative exist. The applicant should provide justification that no other alternatives exist for the SBWR control rod drive (CRD) system. In addition, Figures 4.6-1 and 4.6-2 of the SSAR should be revised to show the individual assemblies described in Paragraph 4.5.1.1 of the SSAR.

SSAR Paragraph 4.5.1.1, "Materials Specifications," states that no cold-worked austenitic stainless steels except those with controlled hardness or strain are employed in the CRD system. These controls are acceptable. However, the applicant should also commit to meet the staff position that the yield strength of cold-worked austenitic stainless steel will not exceed 90,000 psi.

GE Response:

The Fine Motion Control Rod Drive (FMCRD) has a number of components designed to provide wear resistance for various moving parts. The following components are designed for wear, utilizing cobalt-based materials:

- Guide Roller (for Ball Spindle) – Stellite No. 3
- Guide Roller Pin (for Ball Spindle) – Haynes Alloy No. 25
- Guide Shaft (renamed spindle head bolt) – Stellite No. 6B
- Guide Shaft Bushing (renamed spindle head bushing) – Stellite No. 12
- Guide Roller (for Buffer) – Stellite No. 3
- Guide Roller Pin (for Buffer) – Haynes Alloy No. 25
- Stop Piston – 316L, Hardsurfaced with Stellite No. 6

• Hollow Piston Head – 316L, Hardsurfaced with Stellite No. 3

• Ball for Check Valve – Haynes Stellite No. 3

All components have been subjected to a number of development tests using these materials. The drives have been in service for many years in the German KWU BWR reactors, utilizing these same materials and alloys. There currently are no proven substitute materials which have been thoroughly developed and tested to replace the above component materials.

In addition, the drive parts are subjected to very low purge water flow which travels through the drive into the vessel. These low flows result in low corrosion rates and negligible erosion of the parts. The cobalt parts are not exposed to the core environment and therefore are not irradiated. Cobalt addition to the overall coolant concentration is felt to be very low from these FMCRD components.

One FMCRD manufacturer is currently investigating other non-cobalt-based materials which could be used for some of the drive components. Future considerations will be given to replacement of the cobalt-based materials used for some of the drive components with non-cobalt-based materials should investigations and development test results for these replacement materials justify their use.

Figure 4.6-2 of the SSAR will be revised to show the individual assemblies described in paragraph 4.5.1.1. of the SSAR. Figure 4.6-1 is a schematic diagram of the drive and will not be changed.

SSAR paragraph 4.5.5.1, "Materials Specifications," will not be changed. GE does not control cold working of stainless steel based on yield strength, which would be very difficult to measure on cold-worked components. Rather, GE controls cold worked parts primarily by solution heat treatment, by surface hardness and other process controls such as minimum bend radius or induced strain. Surface hardness controls are applied after any straightening or any cold forming processes so that the surface is not allowed to exceed a specified hardness.

NO.	MAIN PART NAME
1	SEAL HOUSING
2	DRIVE SHAFT
3	BEARING
4	GLAND PACKING
5	SPRING (FOR GLAND PACKING)
6	SPOOL PIECE
7	BALL SCREW SHAFT
8	BALL NUT AND BALLS
9	GUIDE ROLLER AND PIN (FOR BALL NUT)
10	SPINDLE HEAD BOLT
11	SPINDLE HEAD BUSHING
12	SEPARATION SPRING
13	MAGNET
14	BUFFER SPRING
15	BUFFER SLEEVE
16	GUIDE ROLLER, PIN
17	STOP PISTON
18	PISTON TUBE
19	PISTON HEAD
20	LATCH
21	LATCH SPRING
22	BAYONET COUPLING
23	COMPRESSION ROD
24	GUIDE TUBE
25	OUTER TUBE
26	MIDDLE FLANGE
27	CHECK VALVE BALL
28	RUBBER SEAL
29	O RING SEAL (BETWEEN CRD HOUSING AND CRD)
30	CRD INSTALLATION BOLT
31	UPPER GUIDE (CRD BLOWOUT SUPPORT)

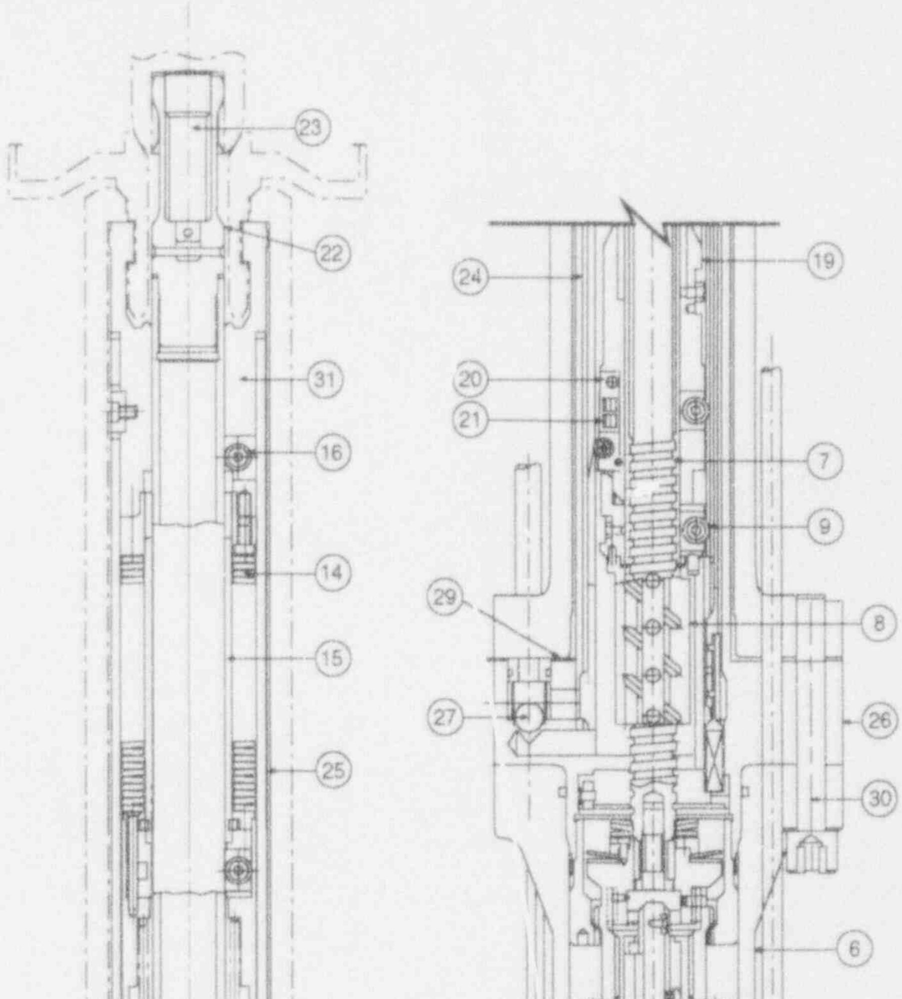


Figure 4.6-2 Fine Motion Control Rod Drive Unit (Cutaway)

RAI Number: EMCB.2

Question:

SSAR Section 5.2.3 Reactor Pressure Boundary Materials

SSAR Paragraph 5.2.4.4.1 states that the SBWR design complies with RG 1.44 and with the guidelines of NUREG-0313. The applicant should commit that the SBWR design complies with NUREG-0313, Revision 2.

The applicant should also commit that cold-worked austenitic stainless steel will conform with the staff position that the yield strength of the steel does not exceed 90,000 psi. SSAR Paragraph 5.2.3.1, "Materials Specifications," must state that the materials for the reactor coolant pressure boundary will be in conformance with the American Society of Mechanical Engineers (ASME) Code, Section III.

SSAR Paragraph 5.2.3.2.2 "BWR Chemistry of the Reactor Coolant," states that hydrogen water chemistry will be used for the SBWR. The applicant must commit to meet the guidelines of RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors;" EPRI's NP-4947-SR, "Hydrogen Water Chemistry Guidelines;" and EPRI's NP-3589-SR-LD, "BWR Water Chemistry Guidelines."

SSAR Paragraph 5.2.3.3.2, "Control of Welding," states that low alloy steel components are either held for an extended time at preheat to ensure removal of hydrogen or preheat is maintained until post-weld heat treatment (PWHT). This approach does not meet regulatory position C2 of RG 1.50 which require that preheat is maintained until PWHT. The staff had previously approved alternative approaches to complying with this requirement. These approaches involved the use of intermittent heating at 400-500°F for 4 hours followed by slow cooling to ambient temperatures or requiring that the component be radiographically examined after final PWHT. The applicant must commit to meet one of these approaches.

SSAR Table 5.2-4, "Reactor Coolant Pressure Boundary Materials," is a list that shows the mounting bolts for the CRD system will be made of SA 194, Grade B7 material. This is an apparent typographical error and should be checked to see if "SA 193, Grade B7" is the correct statement.

GE Response:

These comments are acceptable. The following changes will be made in Amendment 1 of the SSAR (see attached):

Paragraph 5.2.3.4.1 will be revised in Amendment 1 to show compliance with Revision 2 of NUREG-0313.

Paragraph 5.2.3.4.1 (subparagraph Cold-Worked Austenitic Stainless Steels), will include the stipulation that cold-worked austenitic stainless steel will not have a yield stress greater than 90,000 psi.

Paragraph 5.2.3.1, "Materials Specifications," reads as follows: Table 5.2-4 lists the principal pressure retaining materials and the appropriate material specifications for the Reactor Coolant Pressure Boundary (RCPB) components. This statement will also affirm that RCPB materials will conform to the ASME Code, Section III. Do so in the following manner: ... (RCPB) components; all RCPB materials will conform to the American Society of Mechanical Engineers (ASME) Code, Section III.

Paragraph 5.2.3.2.2, "BWR Chemistry of Reactor Coolant," (third paragraph, last sentence) reads thus: Therefore, HWC is used for SBWRs. To specify which hydrogen water controls are to be implemented, extend comment as follows: ... SBWRs; specifically, follow the guidelines set down in RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and EPRI's NP-4947-SR, "Hydrogen Water Chemistry Guidelines," and NP-3589-SR-LD, "BWR Water Chemistry Guidelines."

Paragraph 5.2.3.3.2, "Control of Welding," will meet the regulatory position C2 of RG 1.50 if the fourth sub-paragraph reads thus: All welds are to be nondestructively examined either by radiographic methods (along with a supplemental ultrasonic examination) or by the use of intermittent heating at 400°F to 500°F for four hours, followed by slow cooling to ambient temperatures.

Table 5.2-4, "Reactor Coolant Pressure Boundary Materials," shows that the mounting bolts for the CRD system will be made of SA 194, Grade B7 material. This is a typographical error and should read SA 193, Grade B7.

As a part of the preoperational and startup testing of the main steamlines, movement of the SRV discharge lines will be monitored.

5.2.2.5 Instrumentation Requirements

None.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

Table 5.2-4 lists the principal pressure retaining materials and the appropriate material specifications for the Reactor Coolant Pressure Boundary (RCPB) components; all RCPB materials will conform to the American Society of Mechanical Engineers (ASME) Code, Section III.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 PWR Chemistry of Reactor Coolant

Not applicable to BWRs.

5.2.3.2.2 BWR Chemistry of Reactor Coolant

A brief review of the relationships between water chemistry variables and RCPB materials performance, fuel performance, and plant radiation fields is presented in this section. Further information may be obtained from Reference 5.2-1.

The major environment-related materials performance problem encountered to date in the RCPB of BWRs has been intergranular stress corrosion cracking (IGSCC) of sensitized austenitic stainless steel. IGSCC in sensitized material adjacent to welds in Type 304 and Type 316 stainless steel piping systems has occurred in the past. Substantial research and development programs have been undertaken to understand the IGSCC phenomenon and develop remedial measures. For the SBWR, IGSCC resistance has been achieved through the use of IGSCC resistant materials such as Type 316 Nuclear Grade stainless steel and stabilized nickel-base Alloy 600M and 182M.

However, irradiation-assisted stress corrosion cracking (IASCC) can occur in highly irradiated annealed stainless steel and nickel-base alloys. Preliminary in-reactor and laboratory studies (Reference 5.2-2) have indicated that hydrogen water chemistry (HWC) will be useful in mitigating IASCC. Therefore, HWC is used for SBWRs; specifically, follow the guidelines set down in RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and EPRI's NP-4947-SR, "Hydrogen Water Chemistry Guidelines," and NP-3589-SR-LD, "BWR Water Chemistry Guidelines."

Preheat temperature employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Subsection NA. Components are either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

~~All welds are nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination is performed.~~

All welds are to be nondestructively examined either by radiographic methods (along with a supplemental ultrasonic examination) or by the use of intermittent heating at 400°F to 500°F for four hours, followed by slow cooling to ambient temperatures.

Regulatory Guide 1.34: Control of Electroslag Weld Properties

Electroslag welding is generally not allowed on structural weld joints of low alloy steel.

Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Welder qualification for areas of limited accessibility is discussed under Regulatory Guide 1.71 in Subsection 5.2.3.4.2 of this report.

Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes

Suitable identification, storage, and handling of electrodes, flux, and other welding material will be maintained. Precautions shall be taken to minimize absorption of moisture by electrodes and flux.

5.2.3.3.3 Regulatory Guide 1.66: Nondestructive Examination of Tubular Products

Regulatory Guide 1.66 describes a method of implementing requirements acceptable to NRC regarding nondestructive examination requirements of tubular products used in RCPB. This Regulatory Guide was withdrawn on September 28, 1977, by the NRC because the additional requirements imposed by the guide are satisfied by the ASME Code.

Wrought tubular products are supplied in accordance with applicable ASTM/ASME material specifications. Additionally, the specification for the tubular products which may be used for FMCRD housings specifies ultrasonic examination to Paragraph NB-2550 of ASME Code Section III.

These RCPB components meet 10CFR50 Appendix B requirements and the ASME Code requirements, thus assuring adequate control of quality for the products.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Avoidance of Stress/Corrosion Cracking

Avoidance of Significant Sensitization

When austenitic stainless steels are heated in the temperature range 427 - 982°C, they are considered to become "sensitized" or susceptible to intergranular corrosion. The SBWR design complies with Regulatory Guide 1.44 and with the guidelines of NUREG-0313, to avoid significant sensitization.

Process controls are exercised during all stages of component manufacturing and construction to minimize contaminants. Cleanliness controls are applied prior to any elevated temperature treatment. For applications where stainless steel surfaces are exposed to water at temperatures above 93°C, low carbon (< 0.03%) grade materials are used. For critical applications, nuclear grade materials (carbon content ≤ 0.02%) are used. All materials are supplied in the solution heat treated condition. Special sensitization tests are applied to assure that the material is in the annealed condition.

During fabrication, any heating operation (except welding) between 427 - 982°C are avoided, unless followed by solution heat treatment. During welding, heat input is controlled. The interpass temperature is also controlled. Where practical, shop welds are solution heat treated. In general, weld filler material used for austenitic stainless steel base metals is Type 308L/316L/309L with an average of 8% (of FN) ferrite content.

Process Controls to Minimize Exposure to Contaminants

Exposure to contaminants capable of causing stress/corrosion cracking of austenitic stainless steel components are avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture, construction, and installation.

Special care is exercised to insure removal of surface contaminants prior to any heating operations. Water quality for cleaning, rinsing, flushing, and testing is controlled and monitored. Suitable protective packaging is provided for components to maintain cleanliness during shipping and storage.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guides 1.37 and 1.44.

Cold-Worked Austenitic Stainless Steels

Cold work controls are applied for components made of austenitic stainless steel. During fabrication, cold work is controlled by applying limits in hardness, bend radii and surface finish on ground surfaces. Cold-worked stainless steel will not have a yield stress greater than 90,000 psi.

Table 5.2-4 Reactor Coolant Pressure Boundary Materials (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Head fitting/penetration piping	Forging	Carbon steel	SA 350 LF 2
CRD			
Middle flange	Forging	Stainless steel	SA 182, F304L or 316L
Spool piece	Forging	Stainless steel	SA 182, F304L or 316L
Mounting bolts	Bar	Alloy steel	~~SA 194, B7~~SA 193, Grade B7
Seal housing	Forging	Stainless steel	SA 182, F304L or 316L
Seal housing nut	Bar	Stainless steel	SA 564, Gr 630 (H1100)
Reactor Pressure Vessel			
Shells and Heads	Plate	Mn-1/2 Mo-1/2 Ni	SA-533, Grade B, Class 1
	Forging	3/4 Ni-1/2 Mo-Cr-V Low alloy steel	SA-508, Class 3
Shell and Head Flange	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508 Class 3
Nozzles	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508 Class 3
Drain Nozzles	Forging	3/4Ni-1/2Mo-Cr-V Carbon steel	SA-508 Class 1
Instrumentation Nozzles	Forging	Cr-Ni-Mo Stainless steel	SA-182, Type F316L* or SA-336, Class F8 or F8M or SB-166, SB-167
	Bar, Smls. Pipes	Ni-Cr-Fe	
Stub Tubes	Forging	Ni-Cr-Fe	SB-564, Grade N06600
Isolation Condenser**			
Steam pipe	Seamless	Carbon steel	SA333, Grade 6
Condensate pipe	Seamless	Stainless steel	Type 316L*
Feedwater Piping			
Pipe	Seamless	Carbon steel	SA 333, Grade 6

* Carbon content is not to exceed 0.020%.

** This includes only RCPB materials up to second isolation valve.

RAI Number: EMCB.3

Question:

SSAR Section 5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Boundary

Section 5.2.4 of the SSAR states that the design to perform preservice inspection is based on the requirements of the ASME Code, Section XI, 1989 Edition. The development of the preservice and inservice inspection program plans will be the responsibility of the COL applicant and will be based on the ASME Code, Section XI, Edition and Addenda specified in accordance with 10 CFR Part 50, Section 50.55a. For design certification, GE is responsible for designing the reactor pressure vessel for accessibility to perform preservice and inservice inspection. Responsibility for designing other components for preservice and inservice inspection is the responsibility of the COL applicant. The COL applicant will be responsible for specifying the Edition of the ASME Code, Section XI, to be used, based on the procurement date of the component per 10 CFR Part 50, Section 50.55a. The ASME Code requirements discussed in this section are provided for information and are based on the 1989 Edition of ASME Section XI."

The 1989 Edition of ASME Section XI is referenced in 10 CFR 50.55a(b). Therefore, this national standard is acceptable for use for the preservice inspection (PSI) pursuant to the requirements of 10 CFR 50.55a(g).

The concept of designing the components to perform the preservice inspection based on the 1989 Edition of ASME Section XI is a reasonable approach. However, the staff concludes that the COL applicant must resolve any differences between the reference code (the 1989 Edition) and the code edition required by 10 CFR 50.55(g).

SSAR Paragraph 5.2.4.2, "Accessibility," states that "all items within the Class 1 boundary are designed, to the extent practical, to provide access for the examinations required by ASME Section XI, IWB-2500."

Since the preservice inspection requirements are established and known at the time each component is ordered, 10 CFR 50.55a(g) does not have provisions for "relief requests" for impractical examination requirements. ASME Section XI has provisions to use certain shop and field examinations in lieu of the onsite preservice examination. Therefore, the utility-applicant must incorporate plans for NDE during construction in order to meet all access requirements of the regulations.

SSAR Paragraph 5.2.4.2 describes access for examination of the reactor pressure vessel (RPV).

Examination Category B-A of table IWB-2500-1 requires that an inservice inspection be performed on essentially 100 percent of the weld length of all RPV shell welds as indicated in SSAR Paragraph 5.2.4.3.2. The design of the RPV, biological shield wall and vessel insulation incorporates access for examinations from the outside diameter surface. Automated examinations from the inside diameter surface may be required to completely examine the shell welds and to evaluate the origin of reflectors detected during the inservice inspection. Describe access to the RPV welds for ultrasonic examination from the inside diameter surface.

SSAR Paragraph 5.2.4.3.2 "Examination Methods" indicates that the examination techniques will be based on the 1989 Edition of ASME Section XI as supplemented by RG 1.150 for the RPV.

The ASME Section XI indicates that the preservice examination should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent inservice examinations. Improvements in the ultrasonic testing of reactor coolant pressure boundary (RCPB) components will occur in the near future. The ASME has published in ASME Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC has referenced in 10 CFR 50.55a(b) the ASME Section XI edition that includes the published Appendix VII. In addition, the NRC staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, the SSAR should include provisions that ultrasonic testing during the preservice inspection be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3).

GE Response:

(1) RAI: Section 5.2.4: The staff concludes that the COL applicant must resolve any differences between the reference code (1989 Edition) and the code edition required by 10CFR50.55(g).

Response:
Section 5.2.4 has already indicated that COL applicant will use the ASME Code, Section XI, based on the Edition of the code that is specified in 10CFR50, Section 50.55(a) at the time of procurement of the component. The 1989 Edition of ASME Section XI is for information only.

(2) RAI: Section 5.2.4.2: The utility applicant must incorporate plans for NDE during construction in order to meet all access requirements of the regulations.

Response:
Change the first sentence of Section 5.2.4.2 from: "All items within Class 1 boundary are designed, to the extent practicable, to provide access for the examination required by ASME Section XI, IWB-2500."

To: "All items within the Class 1 boundary are designed to provide access for the examinations required by ASME Section XI, IWB-2500" (see attached).

(3) RAI: Section 5.2.4.3.2: Describe access to the RPV welds for ultrasonic examination from the inside diameter surface.

Response:
Add the following sentence to Section 5.2.4.3.2: "The RPV shell weld are designed for 100% accessibility for both preservice and inservice inspection."

So the inspection requirements read as follows (see attached):

The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. RPV shell welds may be examined from the inside or outside diameter surfaces (or a combination of those techniques) using automated ultrasonic examination equipment.

(4) RAI: The SSAR should include provisions that ultrasonic testing during the preservice inspection be performed in accordance with Appendices VII and VIII pursuant to 10CFR 50.55a(g)(3).

Response:
Add a new Section 5.2.4.3.4 to read as follows (see attached):

5.2.4.3.4 Qualification of Personnel and Examination Systems for Ultrasonic Examination.

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII.

- the main steam and feedwater system up to and including the outermost containment isolation valve.

Exclusions

Portions of the system within the reactor coolant pressure boundary, as defined above, that are excluded from the Class 1 boundary in accordance with 10CFR50, Section 50.55a, are as follows:

- those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; and

- components which are or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and one open). Each such open valve is capable of automatic actuation and if the other valve is open its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only.

The description of portions of systems excluded from the reactor coolant pressure boundary does not address Class 1 components exempt from inservice examinations under ASME Code, Section XI, rules. The Class 1 components exempt from inservice examinations are described in ASME Code, Section XI, IWB-1220.

5.2.4.2 Accessibility

All items within the Class 1 boundary are designed~~, to the extent practicable,~~ to provide access for the examinations required by ASME Section XI, IWB-2500. Items such as nozzle-to-vessel welds often have inherent access restrictions when vessel internals are installed. Therefore preservice examination shall be performed on these items prior to installation of internals which would interfere with examination.

Reactor Pressure Vessel Access

Access for examinations of the reactor pressure vessel (RPV) is incorporated into the design of the vessel, biological shield wall and vessel insulation as follows:

RPV Welds — The shield wall and vessel insulation behind the shield wall are spaced away from the RPV outside surface to provide access for remotely operated ultrasonic examination devices as described in Subsection 5.2.4.3. Access for the insertion of automated devices is provided through removable insulation panels and at shield wall hatches in the upper drywell area. Platforms are attached to the biological shield wall to provide access for installation of remotely operated examination devices.

- reducer to elbow;
- tee to tee; and
- pump to valve.

Straight sections of pipe and spool pieces shall be added between fittings. The minimum length of the spool piece has been determined by using the formula $L = 2T + 152$ mm, where L equals the length of the spool piece (not including weld preparation) and T equals the pipe wall thickness.

5.2.4.3 Examination Categories and Methods

5.2.4.3.1 Examination Categories

The examination category of each item is listed in Table 5.2-6, which is provided as an example for the preparation of the preservice and inservice inspection program plans. The items are listed by system and line number where applicable. Table 5.2-6 also states the method of examination for each item. The preservice and inservice examination plans will be supplemented with detailed drawings showing the examination areas, such as Figure 5.2-7.

For the preservice examination, all of the items selected for inservice examination shall be performed once in accordance with ASME Section XI, IWB-2200 with the exception of the examinations specifically excluded by ASME Section XI from preservice requirements, such as VT-3 examination of valve body and pump casing internal surfaces (B-L-2 and B-M-2 examinations categories, respectively) and the visual VT-2 examinations for categories B-E and B-P.

5.2.4.3.2 Examination Methods

Ultrasonic Examination of the Reactor Vessel
Ultrasonic examination for the RPV will be conducted in accordance with the ASME Code, Section XI. The design to perform preservice inspection on the reactor vessel shall be based on the requirements of the ASME Code, Section XI, 1989 Edition. For the required preservice examinations, the reactor vessel shall meet the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. RPV shell welds may be examined from the inside or outside diameter surfaces (or a combination of those techniques) using automated ultrasonic examination equipment. The RPV nozzle-to-shell welds will be 100% accessible for preservice inspection but might have limited areas that will not be accessible from the outer surface for inservice examination techniques; however, the inservice inspection program for the reactor vessel is the responsibility of the COL applicant and any inservice inspection program relief request

The data so recorded shall be compared with the results of subsequent examinations to determine the behavior of the reflector.

5.2.4.3.4 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with industry accepted programs for implementation of ASME Section XI, Appendix VIII.

5.2.4.4 Inspection Intervals

The inservice inspection intervals for the SBWR will conform to Inspection Program B as described in Section XI, IWB-2412. Except where deferral is permitted by Table IWB-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWB-2412-1. Items selected to be examined within the 10-year intervals are described in Table 5.2-6.

5.2.4.5 Evaluation of Examination Results

Examination results will be evaluated in accordance with ASME Section XI, IWB-3000 with repairs based on the requirements of IWA-4000 and IWB-4000. Re-examination shall be conducted in accordance with the requirements of IWA-2200. The recorded results shall meet the acceptance standards specified in IWB-3400.

5.2.4.6 System Leakage and Hydrostatic Pressure Tests

System Leakage Tests

As required by Section XI, IWB-2500 for Category B-P, a system leakage test shall be performed in accordance with IWB-5221 on all Class 1 components and piping within the pressure retaining boundary following each refueling outage. For the purposes of the system leakage test, the pressure retaining boundary is defined in Table IWB-2500-1, Category B-P, Note 1. The system leakage test shall include a VT-2 examination in accordance with IWA-5240. The system leakage test will be conducted approximately at the maximum operating pressure and temperature indicated in the applicable process flow diagram for the system. The system hydrostatic test (described below), when performed is acceptable in lieu of the system leakage test.

Hydrostatic Pressure Tests

As required by Section XI, IWB-2500 for Category B-P, the hydrostatic pressure test shall be performed in accordance with ASME Section IWB-5222 on all Class 1 components and piping within the pressure retaining boundary once during each 10 year inspection interval. For purposes of the hydrostatic pressure test the pressure retaining boundary is defined in Table IWB-2500-1, Category B-P, Note 1. The system hydrostatic test shall include a VT-2 examination in accordance with IWA-5240. For the purposes of

RAI Number: EMCB.4

Question:

SSAR Section 5.3 Reactor Vessel

SSAR Paragraph 5.3.1.2, "Special Procedures Used for Manufacturing and Fabrication," specifies maximum limits on copper, phosphorous and sulfur for base and weld materials in the beltline region. The applicant must also include a maximum limit of 0.05 vanadium for weld materials in the beltline region.

For staff position regarding compliance with the recommendations of RG 1.50, see Section 5.2.3, Question 5.

SSAR Paragraph 5.3.1.6.1, "Compliance with Reactor Vessel Materials Surveillance Program Requirements," states that three capsules are provided to meet the 10 CFR Part 50 Appendix H requirements. The staff finds this commitment not acceptable since the SBWR is designed for a 60-year life. The applicant must commit to provide at least four capsules and require a minimum capsule lead factor of 1.

SSAR Paragraph 5.3.1.8, "Regulatory Guide 1.65," states that the RPV studs, nut, and washer materials will be ultrasonically examined after final heat treatment and prior to treading. The applicant must also commit to surface examine those items using magnetic particle or liquid penetrant examination after final heat treatment and prior to treading.

SSAR Paragraph 5.3.3.2.1, "Summary Description," states that the interior of the RPV is clad with stainless steel weld overlay and the bottom head is clad with Ni-Cr-Fe alloy. The applicant must specify the cladding process used and identify the weld materials by specification and type.

SSAR Paragraph 5.3.3.2.2, "Reactor Vessel Design Data," states that CRD forged stub tubes for the CRD housing are made of ASME SB-564 materials. The applicant must specify which grade of materials will be used. The applicant should also include the material specifications for the RPV drain nozzles and partial penetration instrumentation water level nozzles.

SSAR Paragraph 5.3.4, "COL License Information," should be revised to reflect that the COL applicant is to provide to the NRC staff for review actual PT limits curves for the specific RPV.

GE Response:

a) SSAR Section 5.3.1.2 will be changed to specify a maximum limit of 0.05% Vanadium for weld materials in the beltline region (see attached).

For response to the staff position concerning compliance with RG 1.50, see RAI # EMCB.2, Question 5. SSAR Para. 5.3.1.4 "Regulatory Guide 1.50" will be revised to incorporate the NRC comment (see attached).

b) The SSAR Para 5.3.1.6.4 will be changed to provide four capsules located in the beltline region, with a minimum lead factor of 1 (see attached).

c) The SSAR Para 5.3.1.8 will be changed to require surface examination of studs, nuts and washer materials using magnetic particle or liquid penetrant examination after final heat treatment and prior to threading (see attached).

d) SSAR Para 5.3.3.2.1 will be revised to include the following:

"A variety of welding processes, such as electroslag, submerged arc, manual welding etc. are used for cladding depending upon the location and configuration of the item in the vessel. Cladding in the "as-clad" condition may be acceptable for some deposits made with automatic processes such as submerged arc welding, gas metal arc welding, and electroslag welding. For other processes, particularly where manual welding is employed, some grinding or machining is required. Workmanship samples are prepared for each welding process in the "as-clad" condition and for typically ground surfaces."

"The welding material used for cladding in the shell area is ASME SFA 5.9 or SFA 5.4, type 309L for the first layer and type 308L or 316L for subsequent layers. For the bottom head cladding, the welding material is ASME SFA 5.14, type ERNiCr3." (see attached)

e) SSAR Para 5.3.3.2.2 and Table 5.2-4 will be revised to specify that the stub tube material is ASME SB-564, Grade N06600. The material specifications for the drain nozzles and water level instrumentation nozzles are specified in Table 5.2-4.

f) The SSAR Para. 5.3.4 will be changed to state the COL applicant will provide actual P/T curves for the RPV (see attached).

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel (RPV) and appurtenances are shown in Table 5.2-4, together with the applicable specifications.

The RPV materials comply with the provisions of ASME Section III, Subsection NB and Appendix I, and also meet the requirements of 10CFR50, Appendix G. The RPV materials also meet the additional requirements as explained in the following subsections.

These materials provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful operating experience in reactor services.

5.3.1.2 Special Procedures Used for Manufacturing and Fabrication

The RPV is constructed primarily from low alloy, high strength steel plate and forgings. Plates are ordered to ASME SA-533, TYPE B, Class 1, and forgings to ASME SA-508, Class 3. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels. Materials used in the core beltline region also specify limits of 0.05% maximum copper, ~~and~~ 0.012% maximum phosphorous content in the base materials and a 0.08% maximum copper and 0.012% maximum phosphorous, and 0.05% maximum vanadium content in weld materials. The maximum sulfur content for base material and weld material is 0.01%

Studs, nuts, and washers for the main closure flange are ordered to ASME SA-540, Grade B23 or Grade B24 having minimum yield strength level of 893 MPa (129.5 ksi). The maximum measured ultimate tensile strength of the stud bolting materials shall not exceed 1172 MPa (170 ksi). Welding electrodes for low alloy steel are low hydrogen type ordered to ASME SFA-5.5.

All plate, forgings, and bolting are 100% ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods in accordance with ASME Section III, Subsection NB-2500.

Fracture toughness properties of materials are also measured and controlled in accordance with ASME Section III, Subsection NB-2300.

All fabrication of the RPV is performed in accordance with GE approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates or forgings, whereas flanges and nozzles are made from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME Section III and IX requirements. Weld test samples are required for each procedure used on major vessel full penetration welds. Tensile and impact tests are performed in accordance with ASME Code, Subsection NB-2300 to determine the properties of the base metal, heat-affected zone (HAZ), and weld metal.

Gas Tungsten Arc Welding (GTAW), Gas Metal Arc Welding (GMAW), Shielded Metal Arc Welding (SMAW), and Submerged Arc Welding (SAW) processes are employed. Electroslag welding is not applied for structural welds. Preheat and interpass temperatures employed for welding of low alloy steel meet or exceed the values given in ASME Section III, Subsection NB, and Appendix D. Post-weld heat treatment at 593°C (1099°F) minimum and not exceeding 635°C (1175°F) is applied to all low-alloy steel welds in accordance with ASME Code, Subsection NB-4620.

Radiographic examination ~~is~~and surface examination are performed on all pressure-containing welds in accordance with requirements of ASME Section III, Subsection NB-5320. In addition, all welds are given a supplemental ultrasonic examination in accordance with ASME Section III, Subsection NB-2530.

The materials, fabrication procedures, and testing methods used in the construction of the SBWR pressure vessels meet or exceed requirements of ASME Section III, Class 1 vessels.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the RPV are examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Section III, Subsection NB-5000. In addition, the pressure-retaining welds are ultrasonically examined. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME Section XI, Appendix I. Acceptance standards are equivalent or more restrictive than required by ASME Section XI.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Regulatory Guide 1.31: Control of Stainless Steel Welding

Controls on stainless steel welding are discussed in Subsection 5.2.3.4.2.

Regulatory Guide 1.34: Control of Electroslag Weld Properties

The requirements of this regulatory guide are not applicable to the SBWR vessel, since electroslag welding is not employed in structural welds.

Regulatory Guide 1.43: Control of Stainless Steel Weld Cladding of Low Alloy Steel Components

The RPV is constructed from low alloy steel forgings or plates conforming to SA-508, Class 3 or SA-533, Type B, which are produced to fine grain practice. Therefore, underclad cracking is not a concern, and the requirements of this regulatory guide are not applicable.

Regulatory Guide 1.44: Control of the Use of Sensitized Stainless Steel

Sensitization of stainless steel is controlled by the use of service proven materials and by use of appropriate design and processing steps, including solution heat treatment, corrosion-resistant cladding, control of welding heat input, control of heat treatment during fabrication and control of stresses.

Regulatory Guide 1.50: Control of Preheat Temperature For Welding Low Alloy Steel

Regulatory Guide 1.50 ~~delineates~~provides suggestions for preheat temperature control ~~requirements~~ and welding procedure qualifications supplementing those in ASME Sections III and IX. Except as noted below, Regulatory Guide 1.50 will be followed.

The use of low alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperature employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Subsection NB, and Appendix D. Components are either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

Acceptance Criterion II.3.b(1)(a) of SRP Subsection 5.2.3 for control of preheat temperature requires that minimum and maximum interpass temperatures be specified. While the SBWR control of low hydrogen electrodes to prevent hydrogen cracking (provided in Subsection 5.2.3.3.2) does not explicitly meet this requirement, the SBWR control will assure that cracking of components made from low alloy steels does not occur during fabrication. Further, the SBWR control minimizes the possibility of subsequent cracking resulting from hydrogen being retained in the weldment.

All pressure-retaining welds are nondestructively examined by radiographic surface examination methods. In addition, a supplemental ultrasonic examination is performed.

Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed under Regulatory Guide 1.71 in Subsection 5.2.3.4.2 of this report.

specimens. The capsule loading consists of 12 Charpy V Specimens each of base metal, weld metal, HAZ material, and three tensile specimens each from base metal and weld metal. A set of out-of-reactor beltline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors will be located within the capsules as required by ASTM E-185.

~~Three~~Four capsules are provided as specified in ASTM E185 as required by 10CFR50, Appendix H, since the design life of the vessel is 60 years and predicted transition temperature shift is less than 56°C (133°F) at the inside of the vessel.

The following proposed withdrawal schedule is ~~in accordance with~~extrapolated from ASTM E 185:

- first capsule: after 6 effective full power years;
- second capsule: after ~~15~~20 effective full power years;
- third capsule: with an exposure not to exceed the peak EOL fluence;
- ~~third~~ fourth capsule: schedule determined based on results of first ~~two~~ three capsules per ASTM E-185, Paragraph 7.6.2.

Fracture toughness testing of irradiated capsule specimens will be in accordance with requirements of ASTM E-185 as required by 10CFR50 Appendix H.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsection 4.1.4.5.

5.3.1.6.3 Predicted Irradiation Effects on Beltline Materials

Transition temperature changes and changes in upper shelf energy are calculated in accordance with the rules of Regulatory Guide 1.99. Reference temperatures are established in accordance with 10CFR50 Appendix G, and Subsection NB-2330 of the ASME Code.

Since weld material chemistry and fracture toughness data are not available at this time, the limits in the design document were used to estimate worst case irradiation effects.

These estimates show that the adjusted reference temperature at end of life for the beltline weld and at the inside of the vessel are less than 18°C (64°F) and 13°C (56°F), respectively, and the end-of-life upper shelf energy exceeds 68 J (50 ft-lb).

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment (Appendix H.II B (2))

Surveillance specimen capsules are located at ~~three~~ four azimuths at a common elevation in the core beltline region. A minimum capsule lead factor of 1 is used in determining the locations of the capsules. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding. Since reactor vessel specifications require that all low alloy steel pressure vessel boundary materials be produced to fine grain practice, underclad cracking is of no concern. The capsule holder brackets allow the removal and reinsertion of capsule holders. Although not Code parts, these brackets are designed, fabricated, and analyzed to the requirements of ASME Section III. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel. (See Subsection 5.3.4 for COL license information requirements pertaining to materials and surveillance capsules.)

In areas where brackets (such as the surveillance specimen holder brackets) are located, additional nondestructive examinations are performed on the vessel base metal and stainless steel weld deposited cladding or weld buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of width equal to at least half the thickness of the part joined. The required stainless steel weld deposited cladding is similarly examined. The full penetration welds are liquid penetrant examined. Cladding thickness is required to be at least 3.2 mm (0.125 in.). These requirements have been successfully applied to a variety of bracket designs which are attached to weld deposited stainless steel cladding or weld buildups in many operating BWRs.

In-service inspection examinations of core beltline pressure retaining welds are performed from either the inside or the outside surface of the RPV. If a bracket for mechanically retaining surveillance specimen capsule holders were located at or adjacent to a vessel shell weld, it would not interfere with the straight beam or half node, angle beam in-service inspection ultrasonic examinations performed from the outside surface of the vessel.

5.3.1.6.5 Time and Number of Dosimetry Measurements

GE provides a separate neutron dosimeter so that fluence measurements may be made at the vessel ID during the first fuel cycle to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence to thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between

2580, after final heat treatment and prior to threading as ~~specified~~ specified.* The examination was in accordance with the requirements of ASME Code, Section II, ASME SA-388 ~~and ASTM A614~~(ASTM A388). The procedures approved for use in practice are judged to insure comparable material quality and are considered adequate on the basis of compliance with the applicable requirements of ASME Code Subsection NB-2580.

The straight beam examination is performed on 100% of cylindrical surfaces and from both ends of each stud using a 19 mm (0.75 in.) maximum diameter transducer. In addition, shear wave examination is also performed from the stud bore using a 60° shear wave probe. The reference standard for the radial scan contains a 12.7 mm (0.5 in.) diameter flat bottom hole with a depth of 10% of the thickness. The end scan standard is per ~~ASTM A614~~ASTM A388. Surface examinations are performed on the studs and nuts after final heat treatment and threading as specified in the guide, in accordance with ~~ASTM A614~~ASTM A388. Any indication greater than that from the applicable calibration feature is unacceptable. The distance/amplitude correction curve for the straight beam end scan of RPV head studs, nuts, and washers is established as follows:

- For studs having a length (L) to O.D. ratio of 7 of less, the distance/amplitude curve is established by a minimum of three test points along the test distance.
- For studs having length to O.D. ratios larger than 7, the minimum number of test points is four. The test points are nearly equally spaced along the test distance. One calibration hole is located at a test distance equal to L/2.

5.3.2 Pressure/Temperature Limits

5.3.2.1 Limit Curves

The pressure/temperature limit curves in Figure 5.3-1 and Figure 5.3-2 are based on the requirements of 10CFR50 Appendix G and Regulatory Guide 1.99.

The vessel flange, RPV head and flange areas, feedwater nozzles, and the core beltline areas were evaluated, and the operating limit curves are based on the most limiting locations. The pressure/temperature limits are based on flaw sizes specified in paragraph 5.3.1.5 (6). The maximum throughwall temperature gradient from continuous heating or cooling at 55.0°C (100°F) per hour was considered. The safety factors applied were as specified in ASME Section III, Appendix G.

The materials for the vessel are provided with the following requirements of RT_{NDT} as determined in accordance with the ASME Section III, Subsection NB-2320: shell and flanges -20°C (-4°F); nozzles - 20°C (-4°F) and beltline welds - 40°C (-40°F).

* The vessel stud, nut, and washer materials are also surface examined using magnetic particle or liquid penetrant examination after heat treatment and prior to threading.

Temperature Limits for Boltup

Minimum flange and fastener temperatures of RT_{NDT} plus 33°C (60°F) are required for tensioning at preload condition and during detensioning. Thus, the minimum limit is -20°C (-4°F) + 33°C (60°F) = 13°C (56°F). This is higher than that calculated in accordance with the methods described in ASME Section III, Appendix G.

Temperature Limits for ISI Hydrostatic and Leak Pressure Tests

Pressure versus temperature limits for preservice and inservice tests when the core is not critical are shown in Figure 5.3-1.

Operating Limits During Heatup, Cooldown, and Core Operation

Figure 5.3-2 specifies limits applicable for reactor operation whenever the core is critical, except for low-level physics tests.

Reactor Vessel Annealing

In-place annealing of the reactor vessel, because of radiation embrittlement, is not necessary because the predicted value of adjusted RT_{NDT} does not exceed 93°C (200°F), as required by 10CFR50 Appendix G, Paragraph IVB.

Predicted Shift in RT_{NDT} and Drop in Upper-Shelf Energy

For design purposes, the adjusted reference nil ductility temperature and drop in the USE for the SBWR vessel is predicted in accordance with the requirements of Regulatory Guide 1.99.

The calculations are based on the specified limits on phosphorous (0.012%), vanadium (0.05%), copper (0.08%) and nickel (1.2%) in the weld material. In plate material and forgings, the limits are copper(0.05%), phosphorous(0.012%)and nickel (1.0%).

The estimated fluence ~~at~~for the ~~beltline~~ weld above the TAF (at the inside of the RPV) and at the support skirt flange (being in the beltline region) are ~~$1.41*10^{18}$ m/cm2~~ $1.41*10^{18}$ n/cm2 and ~~$6.2*10^{17}$ m/cm2~~$6.2*10^{17}$ n/cm2 respectively.

As required by 10CFR50 Appendix H, a surveillance program will be conducted in accordance with the requirements of ASTM E-185. In addition, the specimens located in the beltline forging shall be tested in accordance with the methods of ASTM E-813. The surveillance program will include samples of base metal, weld metal and HAZ material of the beltline forging and also, at the support skirt flange because of its unique position. Subsection 5.3.1.6 provides additional detail on the surveillance program.

5.3.2.2 Operating Procedures

A comparison of the pressure versus temperature limit in Subsection 5.3.2.1 with intended normal operation procedures of the most severe service level B transient shows that those limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures are established so that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the service level B condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas yields a minimum fluid temperature of 276°C (524°F) and a maximum peak gauge pressure of ~~8.38 MPa (1215 psig)~~8.1 MPa (1175 psig). Scram automatically occurs as a result of this event prior to a possible reduction in fluid temperature to ~~121°C (250°F)~~97°C (207°F) at a gauge pressure of ~~6.42 MPa (931 psig)~~5.9 MPa (856 psig). Per Figure 5.3-2, both the ~~8.38 MPa (1215 psig)~~8.1 MPa (1175 psig) vessel gauge pressure at 276°C (529°F) and the ~~6.42 MPa (931 psig)~~5.9 MPa (856 psig) at ~~121°C (250°F)~~97°C (207°F) are within the calculated margin against nonductile failure.

5.3.3 Reactor Vessel Integrity

The reactor vessel materials, equipment, and services associated with the reactor vessel and appurtenances conform to the requirements of the subject design documents. Measures to ensure conformance include (1) provisions for source evaluation and selection, (2) objective evidence of quality furnished, (3) inspection at the vendor source and (4) examination of the completed reactor vessels.

GE provides inspection surveillance of the reactor vessel fabricator in-process manufacturing, fabrication, and testing operations in accordance with the GE quality assurance program and approved inspection procedures. The reactor vessel fabricator is responsible for the first level inspection of manufacturing, fabrication, and testing activities, and GE is responsible for the first level of audit and surveillance inspection.

Adequate documentary evidence that the reactor vessel material, manufacture, testing, and inspection conforms to the specified quality assurance requirements contained in the design documents is available at the fabricator's plant site.

Regulatory Guide 1.2, Thermal Shock to Reactor Pressure Vessels, states that potential RPV brittle fracture, which may result from emergency core cooling system operation, need not be reviewed in individual cases if no significant changes in presently approved core and pressure vessel designs are proposed. If the margin of safety against RPV brittle fracture due to emergency cooling system operation is considered unacceptable, an engineering solution, such as annealing, could be applied to assure adequate recovery of the fracture toughness properties of the vessel material. Regulatory Guide 1.2 requires that engineering solutions be outlined and requires demonstration that the design does not preclude use of the solutions.

An investigation of the structural integrity of boiling water RPVs during a design basis accident (DBA) has been conducted (Reference 5.3-1). It has been determined, based on methods of fracture mechanics, that no failure of the vessel by brittle fracture as a result of DBA will occur.

- expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design and operational limitations assure that NDT temperature shifts are accounted for in reactor operation; and
- operational margins to be observed with regard to the transition temperature are specified for each mode of operation.

Power Generation Design Bases

The power generation design bases of the reactor vessel are:

- develop a simplified system that provides all safety-related functions [i.e., that failure to provide a safety function is incredible (probability of failure is less than 1x10E-6 per year)];
- develop the SBWR vessel with a design life of 60 years with a total plant availability of 87% or greater; and
- design the reactor vessel and appurtenances which allows for a suitable program of inspection and surveillance.

5.3.3.2 Description

5.3.3.2.1 Summary Description

Reactor Vessel

The reactor vessel (Figure 5.3-3) is a vertical, cylindrical pressure vessel of welded low alloy steel forging sections. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with ASME Code, Section III, Class 1 requirements.

In addition, the design documents impose additional requirements to ensure integrity and safety of the vessel. Design of the RPV and its support system meets Seismic Category I equipment requirements. The materials used in the RPV are listed in Table 5.2-4.

The cylindrical shell and top and bottom heads of the RPV are fabricated of low alloy steel, the interior of which is clad with stainless steel weld overlay, except for the head and nozzle weld zones. The bottom head is clad with Ni-Cr-Fe alloy.

<u>A variety of welding processes, such as electroslag, submerged arc, manual welding etc., are used for cladding depending upon the location and configuration of the item in the vessel. Cladding in the "as-clad" condition may be acceptable for some deposits made with automatic processes such as submerged arc welding, gas metal arc welding, and electroslag welding. For other processes, particularly where manual welding is employed, some grinding or machining is required. Workmanship samples are</u>

prepared for each welding process in the "as-clad" condition and for typically ground surfaces.

The welding material used for cladding in the shell area is ASME SFA 5.9 or SFA 5.4, type 309L for the first layer, and type 308L or 316L for subsequent layers. For the bottom head cladding, the welding material is ASME SFA 5.14, type ERNiCr3.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure that design specifications are met.

The vessel head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 55°C (100°F) in any one-hour period. To detect seal failure, a vent tap is located between the two seal rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal ring seal.

Shroud Support

The shroud support brackets (Figure 5.3-3) are welded to the inside of the vessel and are made of Ni-Cr-Fe conforming to ASME SB-168 or SB-166. The shroud brackets support the weight of the steam separators, chimney, core plate and the peripheral fuel bundles. The shroud brackets are classified as core support structures and are designed in accordance with the ASME Section III, Subsection NG.

Protection of Closure Studs

BWRs do not use borated water for reactivity control during normal operation. This subsection is therefore not applicable.

5.3.3.2.2 Reactor Vessel Design Data

The reactor vessel design pressure is 8.62 MPa (1250 psig) and the design temperature is 302°C (576°F). The maximum hydrostatic test pressure is 10.78 MPa (1564 psig).

Vessel Support

The vessel support skirt (Figure 5.3-3) is constructed of low alloy or carbon steel to ASME SA-508, Class 3, SA-516, or SA-533. The top end of the support skirt is welded to the vessel. The vessel support skirt flange is bolted to the steel support structure which is filled with grout. The anchor bolts are set in sleeves which are embedded in the grout. Shear forces are resisted by friction between the skirt flange and the support structure and/or between the flange and anchor bolts. The vessel support skirt is designed to withstand the loading conditions specified in the design documents and meet the stress criteria of ASME Code, Section III, Subsection NF.

Control Rod Drive Housings

The control rod drive (CRD) housings are inserted through the CRD penetrations in the reactor vessel bottom head and are welded to forged stub tubes made of ASME material SB-564 Grade N06600. Each housing transmits loads through the stub tubes to the bottom head of the reactor. These loads include the weights of a control rod, a control rod drive, a control rod guide tube (CRGT), a fuel-support piece (integral with the CRGT), and the four fuel assemblies that rest on the fuel support piece. The housings are fabricated of Type-304 austenitic stainless steel and designed in accordance with ASME Section III, Subsection NB.

In-Core Neutron Flux Monitor Housings

Each in-core neutron flux monitor housing is inserted though the in-core penetrations in the bottom head and welded to forged stub tubes conforming to SB-564.

An in-core flux monitor guide tube is welded to the top of each housing and a startup range neutron monitor (SRNM) or a local power range monitor (LPRM) is bolted to the seal/ring flange at the bottom of the housing outside the vessel. The housings are fabricated of Type-304 austenitic stainless steel and are designed in accordance with ASME Section III, Subsection NB.

Reactor Vessel Insulation

The reactor pressure vessel (RPV) insulation is reflective metal type, constructed entirely of series 300 stainless steel and designed for a 60-year life. The insulation is made up of a combination of two basic shapes— flat panels and cylindrical panels. The insulation for the bottom head and lower shell course inside the vessel support skirt is a vertical cylindrical panel approximately 75 to 100 mm (3 to 4 in.) thick. This panel extends vertically up to the support skirt-to-shell blend radius. There is also a horizontal panel between 75 to 100 mm (3 to 4 in.) thick which connects across the bottom of the vertical insulation panels. This panel is penetrated by the CRD housings, in-core housings, and drain nozzles. These components are not insulated individually.

The insulation for the RPV is supported from the biological shield wall surrounding the vessel and not from the vessel shell. Insulation for the upper head and flange is supported by a steel frame independent of the vessel.

At operating conditions, the insulation on the shield wall and around the refueling bellows has an average maximum heat transfer rate of 176 kcal/m^2h (64.9 Btu/ft^2h) of outside insulation surface. The maximum heat transfer rate for insulation on the top head is 163 kcal/m^2h (60.1 Btu/ft^2h). Minimum air temperatures outside the vessel and insulation are as follows:

- 38°C (100°F), below and outside bottom head insulation and inside the vessel support skirt;

5.3.4 COL License Information

Fracture Toughness Data

Fracture toughness data based on the limiting reactor vessel materials will be provided by the COL applicant (Subsection 5.3.1.5). Pressure/temperature limit curves for the RPV will also be provided (Subsection 5.3.2).

Materials and Surveillance Capsule

The following will be identified: (1) specific materials in each surveillance capsule; (2) capsule lead factors; (3) withdrawal schedule for each surveillance capsule; (4) neutron fluence to be received by each capsule at the time of its withdrawal; and, (5) vessel end-of-life peak neutron fluence (Subsection 5.3.1.6.4)

5.3.5 References

5.3-1 An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident (NEDO-10029).

Table 5.2-4 Reactor Coolant Pressure Boundary Materials (Continued)

Component	Form	Material	Specification (ASTM/ASME)
Head fitting/penetration piping	Forging	Carbon steel	SA 350 LF 2
CRD			
Middle flange	Forging	Stainless steel	SA 182, F304L or 316L
Spool piece	Forging	Stainless steel	SA 182, F304L or 316L
Mounting bolts	Bar	Alloy steel	~~SA 184, B7~~SA 193, Grade B7
Seal housing	Forging	Stainless steel	SA 182, F304L or 316L
Seal housing nut	Bar	Stainless steel	SA 564, Gr 630 (H1100)
Reactor Pressure Vessel			
Shells and Heads	Plate	Mn-1/2 Mo-1/2 Ni	SA-533, Grade B, Class 1
	Forging	3/4 Ni-1/2 Mo-Cr-V Low alloy steel	SA-508, Class 3
Shell and Head Flange	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508 Class 3
Nozzles	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508 Class 3
Drain Nozzles	Forging	3/4Ni-1/2Mo-Cr-V Carbon steel	SA-508 Class 1
Instrumentation Nozzles	Forging	Cr-Ni-Mo Stainless steel	SA-182, Type F316L* or SA-336, Class F8 or F8M or SB-166, SB-167
	Bar, Smls. Pipes	Ni-Cr-Fe	
Stub Tubes	Forging	Ni-Cr-Fe	SB-564, Grade N06600
Isolation Condenser**			
Steam pipe	Seamless	Carbon steel	SA333, Grade 6
Condensate pipe	Seamless	Stainless steel	Type 316L*
Feedwater Piping			
Pipe	Seamless	Carbon steel	SA 333, Grade 6

* Carbon content is not to exceed 0.020%.

** This includes only RCPB materials up to second isolation valve.

RAI Number: EMCB.5

Question:

SSAR Section 5.4.8 Reactor Water Cleanup/Shutdown Cooling System

There should be a provision for automatically maintaining flow through filter/demineralizer units in the event system flow decreases to a point where the bed may drop from septum.

The SSAR should address spent resin transfer from the demineralizers including a description of the monitoring system.

The SSAR should describe the provisions for venting the RWCU components during drain and fill operations.

GE should more explicitly specify materials of construction of the RWCU System. From the statement made in the SSAR, it is not clear if stainless steel is used in the whole RWCU/SDC system or in its water cleanup portion only.

SSAR, page 5.4-29, 3rd paragraph. It should be 56°C/hr (100°F/hr) cooldown rate, instead of 56°C (100°F).

GE Response:

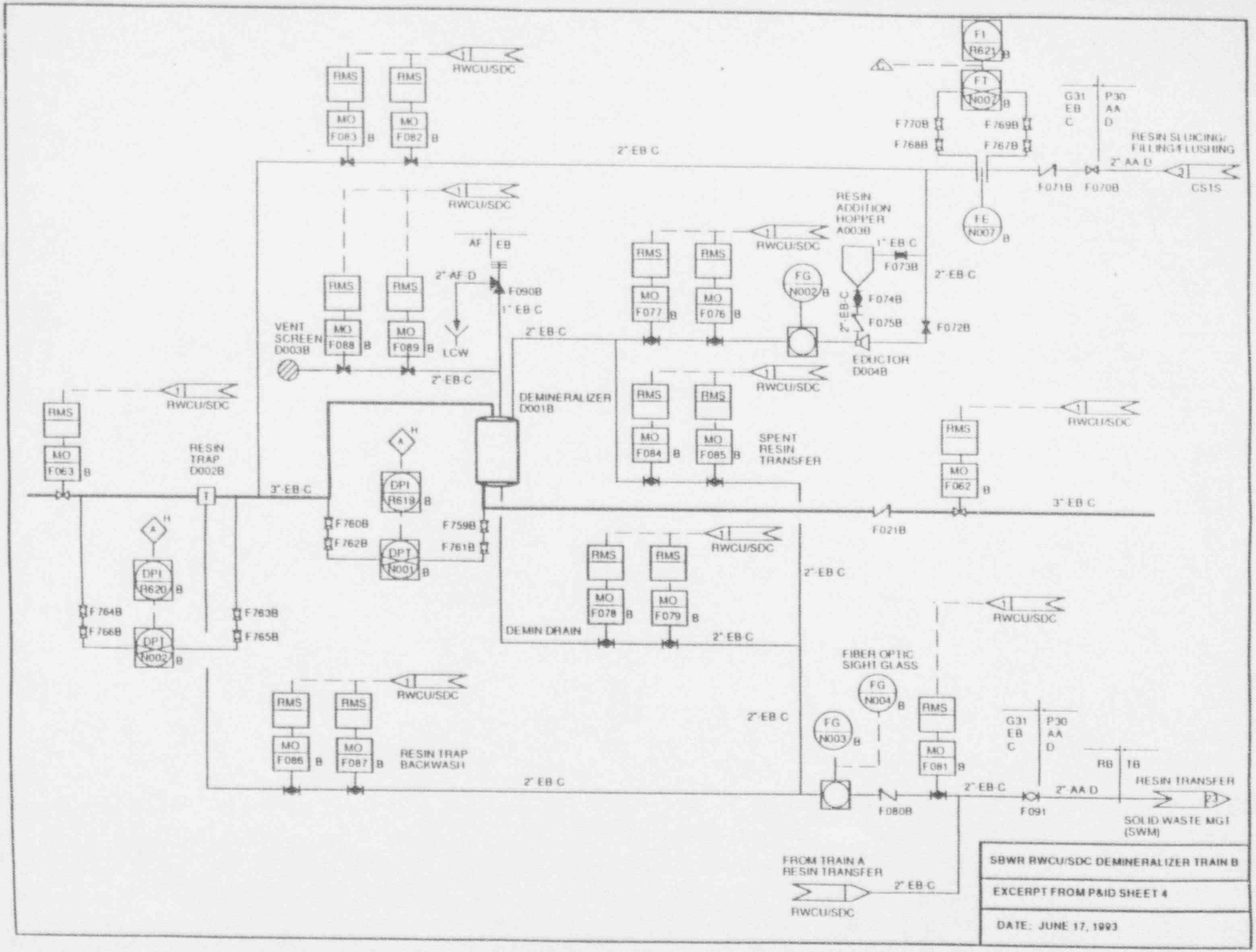
The RWCU/SDC demineralizers are not of the powdered resin filter/demineralizer type which requires provision for automatically maintaining flow through them to hold the resin in place. As explained in Section 5.4.8.1.2 under "Demineralizer," the system has radial flow mixed bed demineralizers. Also, Figure 21.5.4-2, sheet 4 shows that mixed beds are provided. (See attached excerpts from P&ID Sheet 4, dated June 17, 1993.)

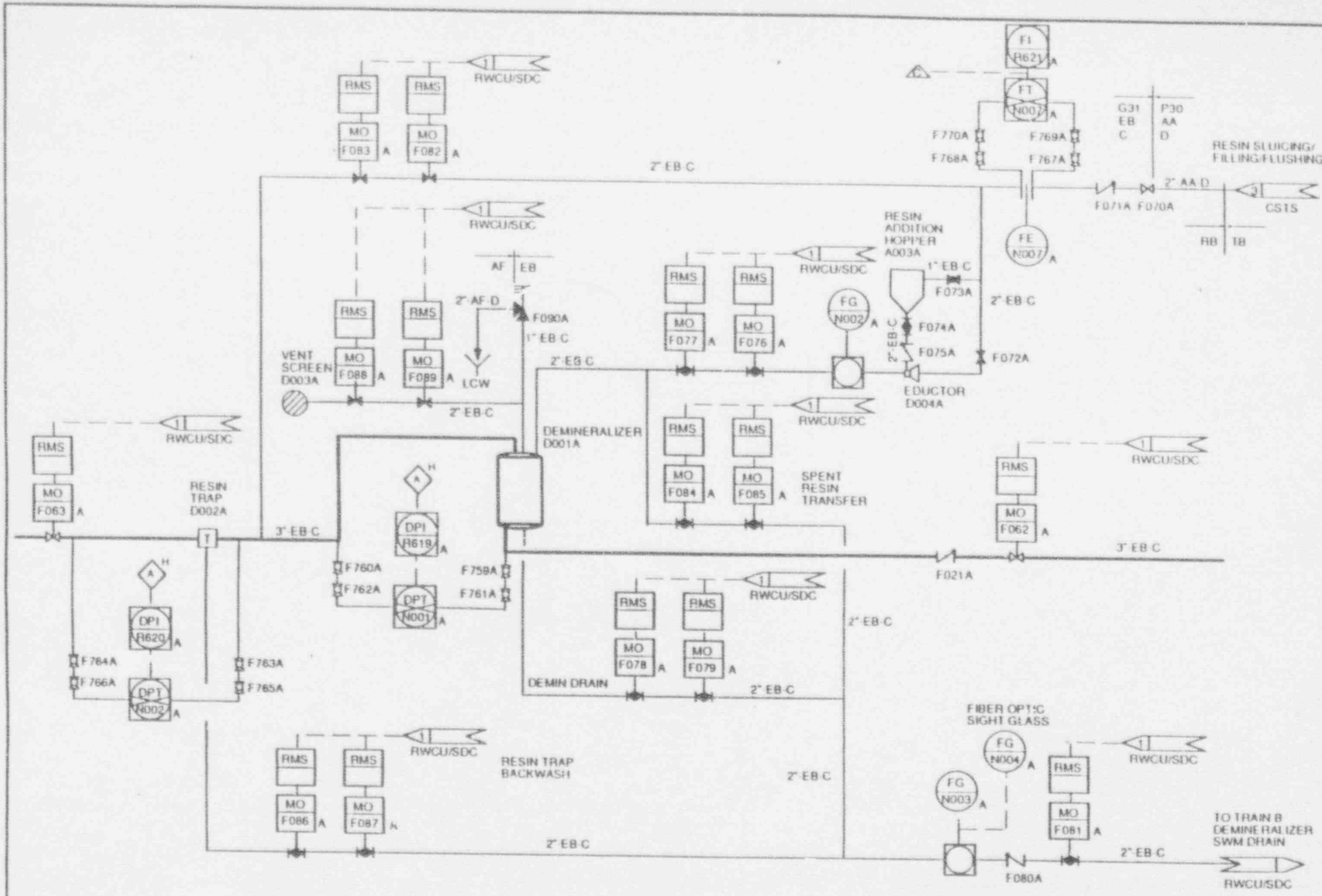
The method for spent resin transfer will be provided in Amendment 1 of the SSAR (see attached). The monitoring system for the demineralizers is discussed in Subsection 9.3.2.

Venting of RWCU components during fill and drain operations will be discussed in Amendment 1 of the SSAR. (see attached)

The statement concerning system materials will be clarified in Amendment 1 of the SSAR to state that all components are stainless steel except for the nonregenerative heat exchanger shell, which is carbon steel (see attached).

The typo concerning cooldown rate will be corrected in Amendment 1 of the SSAR (see attached).


RMS
MO
F083 B
F082 B
RWCU/SDC
2" EB C
FI
R621 B
FT
N007 B
F770B
F768B
F769B
F767B
G31
EB
C
P30
AA
D
RESIN SLUICING/
FILLING/FLUSHING
2" AA D
F071B
F070B
CSTS
FE
N007 B
AF
EB
2" AF D
F090B
1" EB C
LCW
F088 B
F089 B
VENT
SCREEN
D003B
2" EB C
RESIN
ADDITION
HOPPER
A003B
1" EB C
F073B
F074B
F075B
F072B
EDUCTOR
D004B
FG
N002 B
F077 B
F076 B
F084 B
F085 B
SPENT
RESIN
TRANSFER
DEMINERALIZER
D001B
RESIN
TRAP
D002B
F063 B
3" EB C
DPI
R619 B
DPT
N001 B
F760B
F762B
F759B
F761B
F021B
F062 B
DPI
R620 B
DPT
N002 B
F764B
F766B
F763B
F765B
F078 B
F079 B
DEMIN DRAIN
FIBER OPTIC
SIGHT GLASS
FG
N004 B
FG
N003 B
F081 B
F086 B
F087 B
RESIN TRAP
BACKWASH
F080B
F091
RB
TB
RESIN TRANSFER
SOLID WASTE MGT
(SWM)
FROM TRAIN A
RESIN TRANSFER
SBWR RWCU/SDC DEMINERALIZER TRAIN B
EXCERPT FROM P&ID SHEET 4
DATE: JUNE 17, 1993



SBWR RWCU/SDC DEMINERALIZER TRAIN A

EXCERPT FROM P&ID SHEET 4

DATE: JUNE 17, 1993

Each train of the RWCU/SDC System performs the two functions of reactor water cleanup and shutdown cooling with a common piping system. Those portions of the system required for reactor water cleanup are designed for a flow rate of 39 m^3/ hr (172 gpm). ~~The system is all stainless steel with a 3 inch diameter pipe for the reactor water cleanup function.~~ The RWCU/SDC system is constructed of stainless steel except the nonregenerative heat exchanger shell which is carbon steel.

During reactor startup, while maintaining the flow within the cooling capacity of the NRHX, the flow from the demineralizers can be directed to the main condenser hotwell or the liquid radwaste system low conductivity tank for the removal of reactor water that thermally expands during heatup and for removal of inflow from the Control Rod Drive (CRD) System to the RPV.

For RPV hydrotesting and startup, external heating of the reactor water is required if decay heat is not available or the heatup rate from decay heat would be too slow. An electric heater in each train is used to heat the reactor water.

System Components

The supply side of the RWCU/SDC System is designed for the RCPB design pressure plus 10%. Downstream of the pumps, the pump shutoff head at 5% overspeed is added to the supply side design pressure. The RWCU/SDC System includes the following major components:

- demineralizers;
- pumps and adjustable speed motor drives;
- non-regenerative heat exchangers;
- regenerative heat exchangers;
- valves and piping; and
- electrical heater units.

Demineralizer — The RWCU/SDC System has a radial flow mixed bed demineralizer with a low pressure drop design. Design data for the demineralizers is listed in Table 5.4-1.

A full shutdown flow bypass line with a throttle control valve is provided around each demineralizer unit for bypassing these units whenever necessary.

Resin breakthrough to the reactor is prevented by a strainer in the demineralizer outlet line to catch the resin beads. Non-regeneration type resin beads are used, minimizing the potential for damaged beads passing through the strainer to the reactor. The demineralizer is protected from high pressure differential by a bypass valve. The demineralizer is protected from overtemperature by automatic controls that first open the demineralizer bypass valve and then close the demineralizer inlet valve.

Resin bed performance is monitored as described in Subsection 9.3.2. When it is desired to replace the resin, the vessel is isolated from the rest of the system and the resin is sluiced using water to a resin receiving tank. To fill the vessel, fresh resin is added as a slurry. When sufficient resin has been added, the vessel is filled with water, vented, and returned to service.

Pumps — The RWCU/SDC pumps overcome piping and equipment head losses and feedwater line back pressure and return the treated water to the reactor through the feedwater lines.

Pump design data is listed in Table 5.4-2. Figure 5.4-3 shows the expected pump performance for the various operating conditions. The pumps are sealless canned motor (wet stator) type, having zero leakage. Figure 5.4-4 shows a typical cross-section of this type of pump. Drain lines to radwaste are provided to facilitate pump maintenance. A continuous seal purge flow taken from the CRD System is provided to each pump motor. Cooling water for the motor heat exchanger is provided by the Reactor Component Cooling Water System (RCCWS).

Pumps are protected from damage by foreign objects during initial startup by temporary startup suction strainers.

To ensure each pump does not operate against a completely closed discharge, a low flow bypass line is provided around each pump discharge control valve, located upstream of the demineralizer.

Adjustable Speed Drive (ASD) — The RWCU/SDC pumps are each powered from solid-state frequency-converter type ASDs. The ASDs receive 480 V electrical power at constant ac voltage and frequency. The ASDs convert this to a variable frequency and voltage in accordance with a demand signal from a system control unit. The variable frequency and voltage is supplied to vary the speed of the pump motor. The ASD allows effective control of cooldown rate, and reactor temperature after cooldown without the need for throttle valves or cycling the system.

Regenerative Heat Exchanger— Heat exchanger design data for the RHXs is listed in Table 5.4-8. Each RHX is used to recover sensible heat in the reactor water and to reduce the recycle heat loss and avoid excessive thermal stresses and thermal cycles of the feedwater piping. Thermal relief valves are provided on both the shell and tube sides of the RHX.

Startup — During drain and fill operations, the RWCU/SDC system is isolated and depressurized. During draining, the high point vents and low point drains are manually opened. During filling, the low point drains are manually closed and the system is filled with water. Individual high point vents are manually opened to remove any entrapped air.

During heatup, the RWCU/SDC System raises the RPV temperature to 80°C (145°F) using electric heaters provided in each train.

The system is designed to provide sufficient flow, 91 m^3/hr (400 gpm) through the bottom head connections during heatup, cooldown, and startup operations to prevent thermal stratification and to prevent crud accumulation.

During reactor startup it is necessary to remove the CRD purge water injected into the RPV and also the excess reactor water volume arising from thermal expansion. The RWCU/SDC System accomplishes these volume removals and thereby maintains proper reactor level until steam can be sent to the main turbine condenser.

After warmup to approximately 54°C (130°F), the RPV pressure is brought to saturation by opening the vessel to the main condenser through the main steam and turbine bypass lines to promote deaeration of the reactor water. The RWCU/SDC flow rate is reduced to 34 m^3/hr (150 gpm) to prevent cavitation in the piping system and in the pump suction.

The RWCU/SDC System normally removes excess water by dumping, or overboarding, to the condenser hotwell. If the demineralizer is bypassed, the radwaste system is used as an alternative flow path to avoid radioactive coolant from entering the condensate system. Both demineralizer units will be in service, each operating with about two-thirds of the recirculating flow bypassed around the demineralizer unit. While the reactor temperatures are below 71°C (160°F) during the initial startup, about 78 m^3/hr (344 gpm) will be overboarded to the main condenser. Later, when the reactor water temperature exceeds 71°C (160°F), the overboarding flow will be reduced to about 57 m^3/hr (250 gpm), the balance of the process flow is returned to the reactor.

Overboarding is described in more detail below.

Overboarding — During hot standby and startup, water entering the reactor vessel from the CRD System or water level increase due to thermal expansion during plant heatup, may be dumped, or overboarded, to the main condenser to maintain reactor water level.

Overboarding of reactor water is accomplished by using one of the two system trains for overboarding and the other train for the reactor water cleanup function.

The system can be connected to non-safety-related standby ac power (diesel-generators), allowing it to fulfill its reactor cooling functions during conditions when the preferred power is not available.

Using an 8-inch diameter pipe, the shutdown cooling function of the RWCU/SDC System provides decay heat removal capability at normal reactor operating pressure as well as at lower reactor pressures.

The redundant trains of RWCU/SDC permit shutdown cooling even if one train is out of service; however, cooldown time is extended when using only one train.

In the event of loss of preferred power, the system in conjunction with the isolation condensers is capable of bringing the RPV to the cold shutdown condition [≤ 100°C (≤ 212°F)] in 36 hours assuming the most limiting single active failure and with the isolation condensers remove the initial heat load. Refer to Subsection 5.4.8.1.2 for a description of the RWCU/SDC pump motor adjustable speed drive and its operation for shutdown cooling.

System Operation

The modes of operation of the shutdown cooling function are described below:

Normal plant shutdown — The operation of the RWCU/SDC System at high reactor pressure reduces the plant reliance on the main condenser or ICS. The entire cooldown is controlled automatically. As cooldown proceeds and reactor temperatures are reduced, pump speeds are increased and various bypass valves are opened, as described below. During the early phase of shutdown, the RWCU/SDC pumps operate at reduced speed to control the cooldown rate to less than 56°C/hr (100°F/hr) or to less than 14°C/hr (25°F/hr) for the "soft" shutdown procedure.

In order to maintain a ~~56°C (100°F)~~56°C/hr (100°F/hr) cooldown rate, both RWCU/SDC trains are placed into operation early during the cooldown, but with the pumps and system configuration aligned to provide a total system flowrate of approximately 120 m^3/h (530 gpm) with 60 m^3/h (265 gpm) per train. The flow rate for each train is gradually increased to approximately 296 m^3/hr (1300 gpm) thus ending with total system flow rate at a maximum of 592 m^3/h (2600 gpm). To accomplish this, in each train, the bypass line around the RHX, and the bypass line around the demineralizer are opened to obtain this quantity of system flow for the ending condition of the shutdown cooling mode. In addition to the 4-inch RCCWS inlet valve to the NRHX being open, at an appropriate point the 10-inch motor operated RCCWS inlet valve opens to increase the cooling water flow to each NRHX.

The automatic reactor temperature control function controls the ASD controlling the cooldown by gradually increasing the speed of the system pumps up to the maximum pump flow. Water purification operation is continued without interruption.

Notes for Tables 3.3-2a and 3.3-2b:

(1) Windward wall design pressure is positive, side and leeward walls and roof pressures are negative.

(2) For Exposure C:

Add 551 pa (11.5 psf) to windward wall and subtract 551 pa (11.5 psi) from side and leeward walls and roof for Condition I buildings.

Add 551 pa (11.5 psf) to windward wall and subtract 1653 pa (34.5 psf) from side and leeward walls and roof for Condition II buildings.

For Exposure D:

Add 683 pa (14.25 psf) to windward wall and subtract 683 pa (14.25 psi) from side and leeward walls and roof for Condition I buildings.

Add 683 pa (14.25 psf) to windward wall and subtract 2048 pa (42.75 psi) from side and leeward walls and roof for Condition II buildings.

Condition I and Condition II buildings are defined per Table 9 of paragraph 2.2.1a of ASCE Standard 7-1988.

(3) This table is based on the 39.5 meter (130 foot) high Reactor Building, loads must be adjusted for different height buildings.

Table 3.3-3 Factor (qzn/qz130) to Adjust Table 2 Loads for Building Heights Other Than 39.5 Meters (130 ft)

Building Height	(q_z^n/q_z^{130}) Exposure C I = 1.00 and I = 1.11	(q_z^n/q_z^{130}) Exposure D I = 1.00 and I = 1.11
0-15	0.54	0.65
20	0.59	0.69
25	0.63	0.72
30	0.66	0.75
40	0.71	0.79
50	0.76	0.83
60	0.80	0.86
70	0.84	0.89
80	0.89	0.91
90	0.90	0.93
100	0.93	0.95
120	0.98	0.98
130	1.00	1.00
140	1.02	1.02
160	1.06	1.04

RAI Number: EMCB.6

Question:

SSAR Section 6.1 Engineered Safety Feature Materials

Table 6.1-1, "Engineered Safety Features Component Materials," states that containment vessel liner plate may be SA-285 Grade A up to 64mm. This is not acceptable because it does not meet the requirements of Paragraph NE-2221(c), Section III of the ASME Code which require that only fully killed and vacuum degassed steels are used for the containment construction.

GE Response:

This comment is acceptable. The material will be changed in Amendment 1 (see attached) to SA-516 grade 60 or 70 to meet the requirements of ASME Section III, Paragraph NE 2121 (c).

Table 6.1-1 Engineered Safety Features Component Materials

Component	Form	Material	Specification (ASTM/ASME)
Containment[1]			
Containment Vessel Liner[2]			
	Plate ≤64 mm	Carbon Steel	~~SA-285 Gr A~~SA-516 Gr 60 or Gr 70
	Plate >64 mm	Carbon Steel	SA-516 Gr 60 or Gr 70
	Plate	Stainless Steel	SA-240 Type 304L[3]
	Cladding	Stainless Steel	SA-264
Penetrations	Plate	Carbon Steel	SA-516 Gr 60 or Gr 70
			SA-537 Class 1
	Pipe	Carbon Steel	SA-333 Gr 6
Pool Liner	Sheet	Stainless Steel	SA-240 Type 304L or A-167 Type 304L[3]
Drywell Head, Personnel Lock, Equipment Hatch			
	Plate	Carbon Steel	SA-516 Gr 70 or SA-537 Class 1
Structural Steel	Shapes	Carbon Steel	A-36, A572 Gr 50
Vent Pipe	Plate	Stainless Steel	SA-240 Gr 304L
PCCS			
Condenser	Forging	Stainless Steel	SA-182 Gr F304L
	Tube	Stainless Steel	SA-213 Gr TP304L
	Pipe	Stainless Steel	SA-312 Gr TP304L
Piping	Pipe	Stainless Steel	SA312 Gr TP304L
Flanges		Stainless Steel	SA182 Gr F304L
Nuts and Bolts	Bar	Stainless Steel	SA194 Gr 8, SA193 Gr B8

RAI Number: EMCB.7

Question:

SSAR Section 6.1.2 Organic Materials

The SSAR imposes radiation exposure limit for organic materials of 1E10 rads. GE should state that this limit applies for the whole life of the plant.

COL License Information should require that protective coatings in post-accident environments consider the generation of hydrogen from Zn containing primers and topcoats.

The protective coatings should meet the requirements of ANSI 101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities."

GE Response:

As noted in Tables 3D-6, 3D-7, 3D-8, and 3D-9, the integrated dose for normal operation conditions used in equipment qualification means the integrated value over the 60-year life of the plants. In addition, as noted in Tables 3D-14, 3D-15, 3D-16, and 3D-17, the integrated dose for accident conditions means the integrated value over 6 months following a design basis accident. The radiation exposure limits for organic materials are consistent with these integrated doses, the function of the component, and the design life of the component.

Additional information regarding Zn-containing primers and topcoats is not considered necessary. GE does not plan to use any Zn coatings, as none have been qualified for nuclear service to the relevant ANSI standards. Current Zn-rich coatings will not meet EPA regulations as well.

All epoxy coatings will meet the requirements of ANSI N101.2, N101.4, and N5.12, as well as Regulatory Guide 1.54.

RAI Number: EMCB.8

Question:

SSAR Section 6.2.7 COL License Information

SRP 6.2.7, "Fracture Prevention of Containment Boundary," requires that ferritic materials that are part of containment pressure boundary meet the fracture toughness invoked for Class 2 materials effective with Summer 1977 Addenda. The applicant must make this commitment.

GE Response:

As noted in SSAR Table 1.9-1, the SBWR complies with the requirements of SRP Section 6.2.7. However, information describing how compliance is met is currently not provided in Chapter 6 of the SSAR. Therefore, a new SSAR section will be added (Subsection 6.2.7) in Amendment 1 (see attached) that explains how fracture prevention of the containment boundary is assured for SBWR. Other subsections within SSAR Section 6.2 will be renumbered, as deemed appropriate.

Drywell-to-suppression chamber leakage rate tests are performed with the drywell isolated from the suppression chamber. Valves and system lineups are the same as for the ILRT, except for paths that equalize drywell and suppression chamber pressure, which are open during the ILRT and are isolated during the drywell leakage test. The drywell atmosphere is allowed to stabilize for a period of one hour after attaining the test pressure. Leakage rate test calculations, using the suppression chamber pressure rise method, commence after the stabilization period.

The pressure rise method is based on containment atmosphere pressure and temperature observations and the known suppression chamber volume. The leakage rate is calculated from the pressure and temperature data, suppression chamber free air volume, and elapsed time.

Chapter 16 specifies the periodic drywell-to-suppression chamber leakage rate test pressure, duration, frequency, and acceptance criteria.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The reactor containment system includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products attendant postulated accidents.

Fracture prevention of the containment pressure boundary is assured. The SBWR meets the relevant requirements of the following Commission regulations:

- General Design Criterion 1 (as it relates to the quality standards for design and fabrication) — See Subsection 3.1.1.1.
- General Design Criterion 16 (as it relates to the prevention of the release of radioactivity to the environment) — See Subsection 3.1.2.7.
- General Design Criterion 51 (as it relates to the reactor containment pressure boundary design) — See Subsection 3.1.5.2.

~~6.2.7 COL License Information~~

6.2.8 COL License Information

None.

~~6.2.8 References~~

6.2.9 References

6.2-1. W. J. Bilanin, The G.E. Mark III Pressure Suppression Containment Analytical Model, June 1974, (NEDO-20533).

6.2-2. F. J. Moody, Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels, General Electric Company, Report No. NEDO-21052, September, 1975.

6.2-3. Subcompartment Analysis Methods (SCAM). NEDE 2126; 76NED99; Class II

RAI Number: EMCB.9

Question:

SSAR Section 9.3.2 Process and Post Accident Sampling System

The process and post accident sampling System (PASS) should meet the requirements of Section II.B.3 of NUREG-0737.

One of the requirements of NUREG-0737 is that PASS should have capability to analyze liquid samples with the upper nuclide concentration of 10 Ci/g. If the upper limit for measuring PASS samples is only 1 Ci/g, it may take inordinately long time for the samples to decay to this level of activity. In some cases, this decay time may cause unacceptable delay in obtaining the results.

SRP 9.3.2, "Process and Post-Accident Sampling System," requires that, in addition to the sampling described in the SSAR, process sampling system should have capability to take the following samples: sump inside containment, main condenser evacuation system offgas and inlet and outlet of gaseous radwaste storage tank.

GE Response:

Calculations show that reactor water sample radioactivity during post accident conditions is not expected to exceed 1 Ci/g. The reactor water is rapidly diluted with Gravity Driven Cooling System (GDCS) water. If a break occurs in either the reactor vessel or any pipe connected to the reactor vessel, suppression pool water will also dilute the reactor water.

The plant operators will respond to an accident using data from the Safety Parameter Display System, Subsection 18.4.2.11, and will not need reactor water gross activity or isotopic concentration data early in an accident. Samples for these data will be taken later to support accident recovery planning.

Process monitoring for the sump inside containment and the main condenser evacuation system offgas are discussed in Section 11.5.3.2.

There is no gaseous radwaste storage tank. Thus, sampling is not provided.

RAI Number: EMCB.10

Question:

SSAR Section 9.3.9 Hydrogen Water Chemistry

The design of the hydrogen water chemistry system in SBWR should meet the requirements of EPRI Report NP-4500-SR-LD, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations."

GE Response:

SSAR Section 9.3.9.1 indicates that EPRI report NP-4947-SR "BWR Hydrogen Water Chemistry Guidelines" (Reference 9.3-1) is utilized.

RAI EMCB.10 suggests utilization of EPRI report NP-4500-SR-LD "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations." This was a limited distribution document which was ultimately issued as NP-5283-SR-A (same title). We will revise Section 9.3.9.1 in Amendment 1 of the SSAR (see attached) to reference both NP-4947-SR and NP-5283-SR-A for use as appropriate.

9.3.8.5 Instrumentation Requirements

The flow of nitrogen gas from the HPNSS bottled nitrogen is automatically initiated when a pressure switch in the main supply line indicates low nitrogen supply pressure. The pressure switch also alarms in the main control room upon either low or high pressure. The CACS supply isolation valve closes following a set time delay.

9.3.9 Hydrogen Water Chemistry System

9.3.9.1 Design Bases

Safety Design Basis

The Hydrogen Water Chemistry System (HWCS) is non-nuclear, non-safety-related and required to be safe and reliable, consistent with the requirement of using hydrogen gas. The hydrogen piping in the turbine building is designed to Seismic Category II requirements.

Power Generation Design Basis

The BWR reactor coolant is demineralized water, typically containing 100 to 200 parts per billion (ppb) dissolved oxygen from the radiolytic decomposition of water. To mitigate the potential for Intergranular Stress Corrosion Cracking (IGSCC) of sensitized austenitic stainless steels, the dissolved oxygen in the reactor water can be reduced to less than 20 ppb by the addition of hydrogen to the feedwater. The amount of hydrogen required is in the range of 1.0 to 1.5 ppm. The exact amount required depends on many factors including in-core recirculation rates. The amount required will be determined by tests performed during the initial operation of the plant.

The concentration of hydrogen and oxygen in the main steam line and eventually in the main condenser is altered in this process. This leaves an excess of hydrogen in the main condenser that would not have equivalent oxygen to combine with in the offgas system. To maintain the offgas system near its normal operating characteristics, a flow rate of oxygen equal to approximately one-half the injected hydrogen flow rate is injected in the offgas system upstream of the recombiner.

The HWCS utilizes the guidelines given in EPRI report "BWR Hydrogen Water Chemistry Guidelines" (Reference 9.3-1) <u>and "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations" (Reference 9.3-2)</u>.

9.3.9.2 System Description

The HWCS, illustrated in Figure 9.3-1, is composed of hydrogen and oxygen supply systems, systems to inject hydrogen in the feedwater and oxygen in the offgas and several monitoring systems to track the effectiveness of the HWCS. These systems monitor the oxygen levels in the offgas system, the feedwater system, the lower plenum region and the RWCU inlet; hydrogen and pH levels in the feedwater system, the lower

9.3.10.5 Instrumentation

The oxygen supply system has monitors which indicate to the operators when resupply is required. A flow element will indicate the oxygen gas flow rate at all times. The gas flow regulating valves will have position indication in the main control room.

The oxygen monitors are discussed in Subsection 9.3.2.

9.3.11 COL License Information

Hydrogen Water Chemistry System

The COL applicant shall provide an oxygen supply consisting of high pressure gas cylinders or a liquid tank sufficient to meet the requirements of the hydrogen water chemistry system and the oxygen injection system as specified in subsections 9.3.9 and 9.3.10.

9.3.12 References

9.3-1 EPRI Report NP-4947-SR, 1987 Revision, "BWR Hydrogen Water Chemistry Guidelines."

9.3-2 EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations."

RAI Number: EMCB.11

Question:

SSAR Section 9.3.10 Oxygen Injection System

SSAR Paragraph 9.3.10.1, "Design Basis," states that during power operation, deaeration in the main condenser may reduce the condensate oxygen concentration below 20 ppb, thus requiring that oxygen be added. The amount required is up to approximately 5 cubic feet per hour. The last sentence should read: The amount required is up to approximately 5 standard cubic feet per hour.

GE Response:

SSAR Section 9.3.10.1 will be revised in Amendment 1 (see attached) to indicate that the amount of oxygen required is up to approximately 5 standard cubic feet per hour.

concentration below 20 ppb, thus requiring that oxygen be added. The amount required is up to approximately 5 standard cubic feet per hour.

9.3.10.2 System Description

The oxygen supply system will be site dependent and will consist of high pressure gas cylinders or a liquid tank. A condensate oxygen injection module is provided with pressure regulators and associated piping, valves, and controls to depressurize the gaseous oxygen and route it to the condensate injection modules. There are check valves and isolation valves between the condensate injection modules and the condensate lines downstream of the condensate demineralizers and the optional injection point upstream of the filters.

The flow regulating valves in this system are operated from the main control room. The oxygen concentration in the condensate/feedwater system is monitored by analyzers in the Process Sampling System (Subsection 9.3.2). An operator will make changes in the oxygen injection rate in response to changes in the condensate/feedwater concentration. An automatic control system is not required because instantaneous changes in oxygen injection rate are not required.

9.3.10.3 Safety Evaluation

The operation of the Oxygen Injection System is not required to assure any of the following:

- integrity of the reactor coolant pressure boundary;
- capability of shutting down the reactor and maintaining it in a safe shutdown condition; or
- ability to prevent or mitigate the consequences of accidents which can result in potential off-site exposures comparable to the guideline exposure of 10CFR100.

Consequently, the injection system itself is not safety-related. The oxygen storage facility is located in an area in which large amounts of burnable materials are not present. Usual safe practices for handling high pressure gases are followed.

9.3.10.4 Testing and Inspection Requirements

The oxygen injection system is proved operable by its use during normal operation. The system valves may be tested to ensure operability from the main control room. System maintenance can be performed during refueling or maintenance outages.

RAI Number: EMCB.12

Question:

SSAR Section 10.2.3 Turbine Integrity

SSAR Paragraph 10.2.3.1, "Materials Selection," states that Charpy tests will be performed in accordance with ASTM A-170. This is a typographical error and should be corrected to read ASTM A-370.

SSAR Paragraph 10.2.3.4, "Turbine" states that the turbine rotor design will be solid forged monoblock rather than shrunk-on disks. The applicant must specify that the center of the shaft will be bored to remove metal impurities and permit inspection.

GE Response:

SSAR Paragraph 10.2.3.1 will be revised in Amendment 1 of the SSAR (see attached) to correct the typographical error and to state that Charpy tests will be performed in accordance with ASTM A-370.

SSAR Paragraph 10.2.3.4 will *not* be revised to specify that the center of the shaft will be bored to remove metal impurities and permit inspection. The decision whether to bore a hole in the center of a monoblock rotor shaft should be left to the combined operating license (COL) applicant and the rotor vendor. As the manufacturing and inspection technologies continue to improve, the need to bore to remove chemical impurities diminishes. During an ABWR meeting in November 1992, GE informed the NRC that the presence of a bored hole in the shaft introduces additional stresses and reduces the critical flaw size by a factor of 5. Therefore, boring a hole in the center of a monoblock rotor shaft does not *a priori* produce a safer rotor.

of the overspeed protection devices under controlled, overspeed condition is checked at startup and after each refueling or major maintenance outage.

Provisions for testing each of the following devices while the unit is operating are included:

- main stop and control valves;
- turbine bypass valves;
- low pressure turbine combined intermediate valves;
- overspeed governor;
- turbine extraction nonreturn valves;
- condenser vacuum trip system;
- thrust bearing wear detector;
- remote trip solenoids;
- lubricating oil pumps; and
- control fluid pumps.

10.2.3 Turbine Integrity

10.2.3.1 Materials Selection

Turbine rotors and parts are made from vacuum melted or vacuum degassed Ni-Cr-Mo-V alloy steel by processes which minimize flaw occurrence and provide adequate fracture toughness. Tramp elements are controlled to the lowest practical concentrations consistent with good scrap selection and melting practice, and consistent with obtaining adequate initial and long-life fracture toughness for the environment in which the parts operate. The turbine materials have the lowest fracture appearance transition temperatures (FATTs) and hig[illegible] Charpy V-notch energies obtainable, on a consistent basis, from water-quenched [illegible]Cr-Mo-V material at the sizes and strength levels used. Since actual levels of FATT and Charpy V-notch energy vary depending upon the size of the part, and the location within the part, etc., these variations are taken into account in accepting specific forgings for use in turbines for nuclear application. The fracture appearance transition temperature (50% FATT) obtained from Charpy tests performed in accordance with specification ~~ASTM A-470~~ ASTM A-370 will be no higher than -18°C (0°F) for low-pressure turbine disks. The Charpy V-notch energy at the minimum operating temperature of each low-pressure disk in the tangential direction should be at least 81.3 N-m (60 ft-lb).

RAI Number: EMEB.0

Question:

The staff has preliminarily reviewed the critical valves for the SBWR plant design. A crucial part of the review will be the reliability of these types of valves. In order for the staff to begin its review, detailed technical information regarding the design and reliability studies will be needed. The staff recognizes that some of this detailed information may not be available at the design certification stage of the review. However, pertinent assumptions that were used in determining valve design and reliability may be beneficial to the staff and should be substituted in such cases where technical information is unavailable. The staff will also need to conduct working-level meeting(s) with GE to determine if the level of technical information submitted to the staff is adequate in order to complete the SSAR review.

GE Response:

Information Data Sheets have been prepared for those valves considered "critical" for the SBWR plant design. The data sheets include a brief description of the valve, specifications, functional requirements and reliability data and/or assumptions.

The data sheets are attached in response to RAI EMEB.1.

RAI Number: EMEB.1

Question:

Detailed technical information on the design and component reliability are needed for the following list of valves:

- B32 F001, F004 (isolation condenser) N2 rotary 10" and 6" gate MOVs,

- B32 F005 (isolation condenser) NC 6" gate MOV,

- B32 F006 (isolation condenser) NC 6" N2 piston-operated globe valve,

- E50 F002A-F, F006A-C (gravity driven cooling system) 6" squib valves,

- E50 F003A-F, F007A-C (gravity driven cooling system) 6" tilting disc, biased open check valve,

- E50 F009A-C (gravity driven cooling system) 4" squib valve,

- C41 F003A/B (borated water injection line) 2" squib valve,

- SRVs -8 in all (automatic depressurization) dual mode solenoid-operated valve (SOV)/steam-pressure operated, and

- DPVs -F004A-F (automatic depressurization) squib valves.

GE Response:

The attached data sheets have been prepared for the following valves:

B32-F001, F004, F005, F006 — Isolation Condenser System (IC)

C41-F003 — Standby Liquid Control System (SLC)

B21-F006 — Safety / Relief Valve (SRV)

B21-F004, F005 — Depressurization Valve (DPV)

Information for the following Gravity Driven Cooling System (GDCS) valves was transmitted previously on May 14, 1993, MFN No.077-93 :

E50-F002, F006, F009 — Squib Valves
E50-F003, F007 — Biased Open Check Valves

INFORMATION DATA SHEET FOR CRITICAL VALVES

MPL Item No.: B32-F001

System Nomenclature: Isolation Condenser (IC) Isolation Valve

DESCRIPTION

Valve Name/Description

12-inch Nitrogen Rotary Motor Operated Gate Valve (was previously a 10-inch design)

How different from other designs

- Has pneumatic operator to provide diversity of actuator type compared with electric motor operated gate valve in series.
- Double disk or split (parallel) disk design is anticipated for improved operation and performance.

Functional requirements

- The purpose of this valve is to protect against loss of pressure boundary integrity of the outside- containment portion of the Isolation Condenser system.
- The valve closes upon either a signal of excess flow (high Δp) in the IC pipeline or a signal of high radiation in the IC/PCC pool vent line to atmosphere.

Valve and Actuator Characteristics

- 12 inch/minute stem motion;
- Fails as-is on loss of power.

Expected Reliability

(A solenoid valve must open to allow N2 to operate gate valve. Four micro switches operate to control valve operation.)

Failure of valve to close; estimate based on solenoid and electric motor operated valve data (as reported in EPRI URD, Vol. III, Appendix A, "PRA Key Assumptions and Ground Rules):

Failure of solenoid to operate --- 1.0E-3 / demand
Failure of valve to operate (including actuator and switches) --- 4.0E-3 / demand

Failure Rate = 5.0E-3 / demand

VALVE SPECIFICATIONS

Process Fluid Data

- Steam, water, or steam/water mixture can be flashing mixture.
- 1375 psig design pressure, saturation conditions. (110% of reactor system design pressure).

Open/Close Requirements

- Must close against critical flow.
- Transient conditions decreasing from 1250 psig.

Leakage Requirements

- 5 lb/hr air or N_2 with 20 psig across the seated disk, after 40 cycles of operation (closure/exercise) at 1250 psig/50°F to 575°F.

Expected Duty Cycles

- 410 pressurization and heat-up/cool-down cycles without valve operation (0-1250-0 psig; 70°F - 575°F - 70°F saturation).
- 240 full stroke closure/reopen tests at 1050 psig, 550°F.
- 30 closures at $\leq$1250 psig, w/o IC rupture
- 2 closures under IC rupture conditions.

Environment Anticipated

Normal

- Inside containment
- 135°F max, 40%-90% RH, $5x10^6$ rads/year
- Up to 185°F max, local conditions, short duration

Design Accident

- N/A for valve closure (applicable accident is an outside containment break)
- Must remain closed, once closed, for conditions 340°F, decreasing.

Materials

- Carbon steel body.
- Actuator may contain elastomeric seals and nonmetallic electric insulation.

VALVE TESTS

Valve testing

- Engineering type test
- Environmental qualification test
- ASME code tests (e.g., hydrotest)
- Preservice operability test.
- Periodic in-service tests

Code

- Nondestructive examinations
- Hydrostatic testing (ASME Class 1)

Engineering Tests

- Development test needed (low actuator Δp)
- Prototype testing will be performed to confirm the adequacy of design.
- Production tests and inspections will be performed on each unit.

Pre-service

- Valve will be stroked after installation to confirm operability.

EQ

- Operating history (actuator) includes pneumatic supply up to 1500 psi.
- IEEE-323/382 testing required to confirm durability in service/accident. (IEEE Class 1E)

In-service

- Responsibility of Plant Owner
- Operational readiness tests anticipated every 3-months of plant operation.

MPL Item No.: B32-F004

System Nomenclature: Isolation Valve

DESCRIPTION

Valve Name/Description

6-inch Nitrogen Rotary Motor Operated Gate Valve

How different from other designs

- Has pneumatic operator to provide diversity of actuator type compared with electric motor operated gate valve in series.
- Double disk or split (parallel) disk design is anticipated for improved operation and performance.

Functional requirements

- The purpose of this valve is to protect against loss of pressure boundary integrity of the outside- containment portion of the Isolation Condenser system.
- The valve closes upon either a signal of excess flow (high Δp) in the IC pipeline or a signal of high radiation in the IC/PCC pool vent line to atmosphere.

Valve and Actuator Characteristics

- 12 inch/minute stem motion;
- Fails as-is on loss of power.

Expected Reliability

(A solenoid valve must open to allow N2 to operate gate valve. Four micro switches operate to control valve operation.)

Failure of valve to close; estimate based on solenoid and electric motor operated valve data (as reported in EPRI URD, Vol. III, Appendix A, "PRA Key Assumptions and Ground Rules):

Failure of solenoid to operate --- 1.0E-3 / demand
Failure of valve to operate (including actuator and switches) --- 4.0E-3 / demand

Failure Rate = 5.0E-3 / demand

VALVE SPECIFICATIONS

Process Fluid Data

- Saturated water or steam/water mixture. Can be flashing mixture.
- 1375 psig design pressure, saturation conditions. (110% of reactor system design pressure)

Open/Close Requirements

- Must close against critical flow.
- Transient conditions decreasing from 1250 psig.

Leakage Requirements

- 5 lb/hr air or N_2 with 20 psig across the seated disk, after 40 cycles of operation (closure/exercise) at 1250 psig/50°F to 575°F.

Expected Duty Cycles

- 410 pressurization cycles without valve operation, 0-1250-0 psig, 70°F (no thermal cycles)
- 240 full stroke closure/reopen tests at 1050 psig, 550°F.
- 135 thermal cycles, 575°F-70°F-575°F
- 30 closures at $\leq$12.50 psig, w/o IC rupture
- 2 closures under IC rupture conditions.

Environment Anticipated

Normal

- Inside containment
- 135°F max, 40%-90% RH, 5×10^6 rads/year
- Up to 185°F max, local conditions, short duration

Design Accident

- N/A for valve closure (applicable accident is an outside containment break)
- Must remain closed, once closed, for conditions 340°F, decreasing.

Materials

- Stainless steel body.
- Actuator may contain elastomeric seals and nonmetallic electric insulation.

VALVE TESTS

Valve testing

- Engineering type test
- Environmental qualification test
- ASME code tests (e.g., hydrotest)
- Preservice operability test.
- Periodic in-service tests

Code

- Nondestructive examinations
- Hydrostatic testing (ASME Class 1)

Engineering Tests

- Development test needed (low actuator Δp)
- Prototype testing will be performed to confirm the adequacy of design.
- Production tests and inspections will be performed on each unit.

Pre-service

- Valve will be stroked after installation to confirm operability.

EQ

- Operating history (actuator) includes pneumatic supply up to 1500 psi.
- IEEE-323/382 testing required to confirm durability in service/accident. (IEEE Class 1E)

In-service

- Responsibility of Plant Owner
- Operational readiness tests anticipated every 3-months of plant operation.

MPL Item No.: B32-F005

System Nomenclature: Isolation Condenser Condensate Return

DESCRIPTION

Valve Name/Description

6-inch, DC Electric Motor Operated Gate Valve

How different from other designs

- Electric motor operated gate valves are standard design.
- Split (parallel) disk or double disk type gate is anticipated for improved operation and performance.

Functional requirements

Normally closed; one of two IC condensate return valves in parallel; opens upon initiation of the IC system to permit return of condensed reactor steam back to the reactor vessel.

Valve and Actuator Characteristics

- 12 inch/minute stem motion;
- Fails as-is on loss of power.

Expected Reliability

Failure to operate:

NPRDS Data

4" - 11.99" Electric Motor Operated Gate Valve
Data based on past 10 years operation - all plants

Valve: 4641 records ; 4.1E+8 calendar hours ; 292 failures ; 7.1E-7 / hour

Motor Operator: 654 records ; 5.38E+7 calendar hours ; 407 failures ; 7.5E-6 / hr

Failure rate = 8.2E-6 / hour

VALVE SPECIFICATIONS

Process Fluid Data

- Normally subcooled condensate
- Temperature increases to 575°F when valve is first opened
- 1375 psig design pressure saturation conditions. (110% of reactor system design pressure.)

Open/Close Requirements

- Must open against 20 psi differential pressure from 0-1250 psig.

Leakage Requirements

- 5 lb/hr air or N_2 with 20 psig across the seated disk, after 40 cycles of operation (closure/exercise) at 1250 psig, 50°F to 575°F.

Expected Duty Cycles

- 410 pressurization cycles without valve operation (0- 1250 -0 psig); 70°F (no thermal cycles)
- 240 partial stroke openings; pressure differential negligible;
- 135 thermal cycles, 575°F-70°F-575°F

Environment Anticipated

Normal
- Inside containment
- 135°F max, 40%-90% RH, $5x10^6$ rads/year
- Up to 185°F max, local conditions, short duration

Design Accident
- N/A for valve opening (applicable accident initiates valve opening)
- Must remain open, once opened.

Materials

- Stainless steel body.
- Actuator may contain elastomeric seals and nonmetallic electric insulation.

VALVE TESTS

Valve testing

- Environmental qualification test
- ASME code tests (e.g., hydrotest)
- Preservice operability test.
- Periodic in-service tests.

Code

- Nondestructive examinations
- Hydrostatic testing (ASME Class 1)

Engineering Tests

- Prototype testing will be performed to confirm the adequacy of design.
- Production tests and inspections will be performed on each unit.

Pre-service

- Valve will be stroked after installation to confirm operability.

EQ

- EQ (IEEE-323/382) may be by analysis (known hardware, similar conditions to existing service).

In-service

- Responsibility of Plant Owner
- Operational readiness tests anticipated every 3-months of plant operation.

MPL Item No.: B32-F006

System Nomenclature: Isolation Condenser Condensate Return

DESCRIPTION

Valve Name/Description

6-inch, Pneumatic Piston-Operated Globe Valve

How different from other designs

- Uses pneumatic operator to provide diversity of actuator type (mounted in parallel with F005); actuator holds valve closed; valve is opened by spring on shutoff or loss of pneumatic pressure / electrical signal.

Functional requirements

Normally closed; one of two IC condensate return valves in parallel; opens upon initiation of the IC system to permit return of condensed reactor steam back to the reactor vessel.

Valve and Actuator Characteristics

- 12 inch/minute stem motion.
- Fails open on loss of power.

Expected Reliability

(Two in-series solenoid valves must close to isolate from N2 source.)

Failure to operate or solenoid failure to close
NPRDS Data
All sizes of solenoid valves
Data based on past 10 years operation - all plants
4080 records ; 2.73E+8 calendar hours ; 480 failures
Failure Rate = 1.7E-6 / hour ; Two valves = 3.4E-6 / hour

Air operated valve - failure to operate to de-energized position
2.0E-6 / hour (ALWR Requirements Document, Chapter 1, Appendix A*)

Failure Rate = 5.4E-6 / hour

* Note: Summary shows 1.0E-6/hr (typing error), but "Survey 7" survey data correctly shows 2.0E-6/hr.

VALVE SPECIFICATIONS

Process Fluid Data

- Normally subcooled condensate
- Temperature increases to 575°F when valve is first opened
- 1375 psig design pressure saturation conditions. (110% of reactor system design pressure.)

Open/Close Requirements

- Must open against 20 psi differential pressure from 0-1250 psig.

Leakage Requirements

- 5 lb/hr air or N_2 with 20 psig across the seated disk, after 40 cycles of operation (closure/exercise) at 1250 psig, 50°F to 575°F.

Expected Duty Cycles

- 410 pressurization cycles without valve operation (0- 1250 -0 psig; 70°F (no thermal cycles)
- 240 partial stroke openings; pressure differential negligible;
- 135 thermal cycles, 575°F-70°F-575°F

Environment Anticipated

Normal

- Inside containment
- 135°F max, 40%-90% RH, $5x10^6$ rads/year
- Up to 185°F max, local conditions, short duration

Design Accident

- N/A for valve opening (applicable accident initiates valve opening)
- Must remain open, once opened.

Materials

- Stainless steel body.
- Actuator may contain elastomeric seals and nonmetallic electric insulation.

VALVE TESTS

Valve testing

- Environmental qualification test
- ASME code tests (e.g., hydrotest)
- Preservice operability test.
- Periodic in-service tests.

Code

- Nondestructive examinations
- Hydrostatic testing (ASME Class 1)

Engineering Tests

- Prototype testing will be performed to confirm the adequacy of design.
- Production tests and inspections will be performed on each unit.

Pre-service

- Valve will be stroked after installation to confirm operability.

EQ

- EQ (IEEE-323/382) may be by analysis (known hardware, similar conditions to existing service).

In-service

- Responsibility of Plant Owner
- Operational readiness tests anticipated every 3-months of plant operation.

MPL Item No.: C41-F003A & B

System Nomenclature: Standby Liquid Control System Injection Valve

DESCRIPTION

Valve Name/Description

2-inch, nominal, squib-type SLC System Injection Valve

How different from other designs

- Higher design pressure (2500 psig).
- Larger throat area for increased Cv, higher flow rates.

Functional requirements

- Normally closed; opens on signal to permit injection of sodium pentaborate solution for ATWS mitigation.
- May serve isolation function (in closed position).

Valve and Actuator Characteristics

- Squib (explosive) actuated valve.
- Explosive is detonated by a signal to the firing circuit.
- Explosive drives a steel shaft (the "ram") vertically downward.
- Ram shears off a precisely-machined portion of the disk (the "cap").
- Ram is held (wedged) in a conical section to keep out of SLC piping.

Expected Reliability

- Equal to or better than the EPRI URD, Vol. III, Appendix A, "PRA Key Assumptions and Ground rules."
- Failure to open --- 3.0E-3 / demand
  (used in SBWR PRA; same as the EPRI URD value)
- Inadvertent operation --- 5.74E-8 / hr
  (based on 696 years of BWR experience, SLC system, 2 valves / plant, no inadvertent operations. This is superior to the EPRI URD value of 4.0E-7 / hr)

VALVE SPECIFICATIONS

Process Fluid Data

- System design pressure is 2500 psig from accumulator to reactor entry nozzle.
- Pressure of 12.5% concentration sodium pentaborate solution on valve inlet side is 2500 psig
- Normal condition on valve outlet side is demineralized water at reactor pressure (0 - 1250 psig design pressure) when valve is closed.

Open/Close Requirements

- Opens on time-delayed ATWS signal (3 min. delay)
- Fast-opening, <<0.1 sec (actual).
- Does not reclose without refurbishment. Internals (squib, ram, disk) are not reusable; these must be replaced after use.

Leakage Requirements

- Zero leakage (leak tight).

Expected Duty Cycles

- Upstream side is exposed to $\geq$15 depressurization cycles (from 1250 psig to zero) for squib replacement — anticipated every 4 years, 60 year life
- Downstream side is exposed to reactor pressure cycles (pressure only, not temperature).

Environment Anticipated

Normal
- Room temperature (mild environment)
- 0-100% RH

Design Accident
- Approximately 150°
- 0-100% RH

Materials

- Stainless steel body
- Explosive: Diazo dinitrophenol
- Primer: Lead azide

VALVE TESTS

Valve testing

- Engineering tests
- Environmental qualification.
- ASME code tests.
- Operability monitoring.

Code

- Nondestructive examinations
- Hydrostatic testing (ASME Class 1)

Engineering Tests

- Engineering tests (by vendor).
- Production tests and inspections will be performed on each unit.

Pre-service

- Circuit continuity check.
- Bridgewire resistance check

EQ

- EQ tests have been performed on similar designs which have identical elastomeric and explosive materials.
- Seals have similar shape and function.
- EQ may therefore be established by similarity and analysis. (IEEE Class 1E)

In-service

- Operational testing is not possible; instead, the firing circuit is capable of being monitored for circuit continuity.

MPL Item No.: B21-F006 (SRVs)

System Nomenclature: Safety/Relief Valve

DESCRIPTION

Valve Name/Description

8x10 Main Steam Line Direct Acting Safety/Relief Valve
(Over pressure Protection, ADS, and Manual Relief)
(8 inch inlet flange, 10 inch outlet flange)

How different from other designs

- Expected to be the same design as existing SRVs used on currently-operating BWR/6 plants. Direct acting SRV; not pilot operated.
- May add a position indicator, such as a LVDT, for stem position detection

Functional requirements

- Opened by steam pressure, which overcomes spring force to lift the disk (safety mode operation) - over pressure protection function.
- Opened by auxiliary power actuator (relief mode operation) - ADS function and manual relief.

Valve and Actuator Characteristics

- "Servo-air" solenoid-driven pneumatic control valve provides air or N_2 to lifting cylinder which lifts a lever arm to lift the SRV stem.
- Fast relief mode opening time; less than 0.2 seconds.

Expected Reliability

Experience Basis: NPRDS data for direct acting Main Steam SRVs on GE BWRs, last 10 years of operation for Crosby and Dikkers SRV types; 1.07E+7 hours.

- SRV stuck open beyond Tech. Spec. time limit: 2 events ⇒ 1.87E-7 / hr *
- Inadvertent opening in relief (actuator) mode of operation: 3 events ⇒ 2.80E-7 hr *
- Failures to open in relief (actuator) mode of operation: 4 events ⇒ 3.74E-7 / hr *
  (2 - electrical connection; 1 - switch, 1 - unknown)

* This performance is significantly superior to EPRI URD numbers because the EPRI URD numbers are based on pilot-operated SRVs rather than direct acting SRVs.

- SRV Safety mode (spring) setpoint data (found during surveillance/refurbishment) outside the allowable range of ±3 % of nameplate setpoint: 15 events ⇒ 1.40E-6 / hr (12 known; 3 additional events estimated on the basis of incomplete information)
- Largest deviations from setpoint reported in the data base: 5.6% low ; 4.45% high
- Theoretical frequency of failure to open in safety mode, based on the above experience basis and zero catastrophic failures to open reported to date:

Failure Rate = 6.5E-8 / hour

VALVE SPECIFICATIONS

Process Fluid Data

- Normally saturated steam, 99% quality.
- Saturated steam to 50% quality steam/water mixture, accident case(s).
- 1375 psig design pressure, saturation conditions (110% of reactor system design pressure)

Open/Close Requirements

- Safety mode - fully open with full rated capacity discharge at 103% of spring set pressure.
- Relief mode - open upon receipt of manual electrical signal to solenoids.

Leakage Requirements

- Design capability - ≤20 lb/hr (steam) after 180 power (relief) actuation's plus 20 steam (safety) actuation's.

Expected Duty Cycles

- 410 pressurization and heat-up/cool-down cycles without valve operation (0-1250-0 psig; 70°F-575°F-70°F, saturation)
- Approximately 40 full stroke open/close tests during 60 year life (one during startup after each outage).
- Design capability >1000 relief actuations per 5-year maintenance interval (>> expected service)
- ≤2 safety actuations per 60-year plant life.

Environment Anticipated

Normal

- Inside containment
- 135°F max, 40%-90% RH, $5x10^6$ rads/year
- Up to 185°F max, local conditions, short duration

Design Accident

- 340°F, 100% RH for 3 hour
- 303°F, 100% RH for 3 additional hours
- 303°F, 100% RH, 6 hr.- 100 days (subject to change)

Materials

- Carbon steel body

VALVE TESTS

Valve testing

- Engineering type tests.
- Environmental qualification tests.
- ASME code tests.
- Preservice operability test (performed during startup)
- Recertification tests
- Post-outage startup tests

Code

ASME, CLASS 1 AND OVER PRESSURE PROTECTION DEVICE

- Nondestructive examinations.
- Hydrostatic testing
- Capacity certification
- Set point testing and adjustment
- Blowdown adjustment

Engineering Tests

- Will use same as existing designs; development tests not required (except possibly for position indicator).
- Prototype tests complete (except possibly for position indicator).
- Production tests and inspections will be performed on each unit.

Pre-service

- Relief mode operating during startup at $\geq$800 psi - operability test with steam under disk to minimize d‹ ·lopment of steam leakage past the seat.

EQ

- EQ tests have been performed on identical designs at similar or more severe environments (except that position indicator may require qualification).
- IEEE-323/382, NUREG-0588 (IEEE Class 1E)

In-service

- Responsibility of plant owner.
- Should be performed per OM-1. (recertification testing - set point and leakage, actuator refurbishment - on planned interval of 5 years or less per each SRV.

MPL Item No.: B21-F004, F005 (DPVs)

System Nomenclature: Depressurization Valve

DESCRIPTION

Valve Name/Description

12 x 7 x 12 Main Steam System Depressurization (squib) Valve
(12 inch inlet flange; 7 inch orifice; 12 inch outlet flange)

How different from other designs

- New technology
- Significantly larger model of squib operated valve than any previous squib valve model used in GE BWR plants
- Technically a propellant-operated valve (not an explosive-operated valve).

Functional requirements

- Normally closed; opens in response to LOCA/low reactor water level events to depressurize the reactor

Valve and Actuator Characteristics

- Squib (propellant) actuated valve.
- Propellant is ignited by a signal to the ignitor circuit.
- Propellant drives a piston vertically downward.
- Piston shears off a precisely machined portion of the disk ("nipple"), opening the valve.
- Disk is held in place by a hinge pin, so as not to become a projectile.

Expected Reliability

- Significantly better* than the figures given for "explosive valves" by EPRI URD, Vol. III, Appendix A, "PRA Key Assumptions and Ground rules."
- Fail to open: 3.0E-3 / demand (used in SBWR PRA; same as the URD value)
- Inadvertent operation: 5.74E-8 / hr (Based on 696 years of BWR experience, SLC system, 2 valves / plant, no inadvertent operations. This is superior to the URD value of 4.0E-7 / hr)

* "Significantly better", based on the following:
Historical data for reliability of the BWR explosive valves is based on the explosive valves used in the SLC system. These valves use a lead azide primer and diazodinitrophenol squib. The DPV primer uses a zirconium/potassium perchlorate primer and a carbon/potassium nitrate squib.

- From the standpoint of failure to open, the zirconium/potassium perchlorate primer is more likely to fire reliably upon receipt of the electrical signal. (Both squib types will fire reliably when the primer is ignited.)
- From the standpoint of inadvertent operation, both primers are stable. The zirconium / potassium perchlorate squib is much more stable than the diazodinitrophenol squib, and remains stable at much higher temperature and radiation levels

VALVE SPECIFICATIONS

Process Fluid Data

- Normally saturated steam, 99% quality.
- Saturated steam to 50% quality steam/water mixture, accident case(s).
- 1375 psig design pressure, saturation conditions (110% of reactor system design pressure)

Open/Close Requirements

- Opens on signal to the ignitor circuit.
- <0.45 sec (spec); $<<0.1$ sec (actual).
- Does not reclose without refurbishment. Internals (squib, piston, disk) are not reusable; these must be replaced after each use

Leakage Requirements

- Zero leakage (leak tight).

Expected Duty Cycles

- 410 pressurization and heat-up / cool-down cycles without valve operation (0-1250-0 psig; 70°F-575°F-70°F, saturation)
- ≤ 2 operations during 60-year service

Environment Anticipated

Normal

- Inside containment
- 135°F max, 40%-90% RH, 5×10^6 rads/year
- Up to 185°F max, local conditions, short duration

Design Accident

- 340°F, 100% RH for 3 hour
- 303°F, 100% RH for 3 additional hours
- 303°F, 100% RH, 6 hr.–100 days (subject to change)

Materials

- Stainless steel body
- Propellant: Carbon, Potassium Nitrate
- Primer: Zirconium Potassium Perchlorate

VALVE TESTS

Valve testing

- Engineering type tests.
- Environmental qualification tests.
- ASME code tests.
- Operability monitoring

Code

- Non destructive examinations
- Hydrostatic testing (ASME Class 1)

Engineering Tests

- Development tests are completed.
- Prototype tests are complete.
- Production tests and inspections will be performed on each unit.

Pre-service

- Circuit continuity check.
- Bridgewire resistance check.

EQ

- EQ tests have been performed on a prototype unit including multiple primers.
- EQ for production units will be by identity/similarity analysis. (IEEE Class 1E)

In-service

- Operational testing is not possible; instead, the firing circuit is capable of being monitored for circuit continuity.

RAI Number: HHFB.2

Question:

SSAR Section 13.2.1-2, "Training"

The details of the site-specific training program are not within the scope of the SBWR standard design certification. However, the application should provide a description of the process to ensure that technically relevant training information is provided to the COL applicant.

GE Response:

GE agrees that the details of the site-specific training program are not within the scope of the SBWR standard design certification. GE also recognizes the need to provide technically relevant training information beyond the design certification information to the COL applicant. It is GE's intent that the details of the agreement(s) in this area will be established between GE and the applicant sometime during the procurement process.

RAI Number: HHFB.3

Question:

SSAR Section 13.5.2, "Plant Procedures"

The details of the site-specific procedure development program are not within the scope of the SBWR standard design certification. However, the application should provide a description of the process to ensure that technically relevant procedure development information is provided to the COL applicant.

GE Response:

GE agrees that the details of the site-specific procedure development program are not within the scope of the SBWR standard design certification. GE also recognizes the need to provide technically relevant procedure development information beyond the design certification information to the COL applicant. It is GE's intent that the details of the agreement(s) in this area will be established between GE and the applicant sometime during the procurement process.

RAI Number: HHFB.4.2

Question:

DAC for this chapter were not included in the application.

GE Response:

The Design Acceptance Criteria (DAC) for Human Factors Engineering (HFE) were not available at the time of the August 1992 or February 1993 submittals of Chapter 18. However, the Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) were included in the February 1993 submittal. SBWR SSAR Amendment 1 will include the DAC for Chapter 18 based on the resolution of the DAC for ABWR.

RAI Number: HHFB.4.5

Question:

The application does not identify the specific sources of operational experience used to develop the control room standard design features, nor how the lessons learned from such experience were incorporated into the SBWR MMIS design and implementation process described in Appendix 18E.

GE Response:

The standard design features of the SBWR MMIS are identical to those in the ABWR MMIS. The specific sources of operational experience and other information utilized in the development of the standard design features are discussed in Appendix 18G of the ABWR SSAR (GE Document 23A6100).

Appendix 18E is being revised to document the design implementation process in Amendment 1 of the SSAR.

RAI Number: HHFB.4.6

Question:

The application does not identify the methodology used for selection of the design goals, the bases and the criteria used for selection of an individual standard design feature, or why the feature was selected for use.

GE Response:

The design goal and design bases, described in Section 18.2, were formulated through the application of engineering judgement to the problem of creating a truly modern MMIS. A five-year development program, which included surveys of trends in control station design in all industries, research in the fields of automation, display technology, operator work load and many other related areas and questionnaires to many segments of the management and workers in industry, contributed information which was utilized in formulating the design goal and design bases of the ABWR and SBWR MMIS.

The selection of the individual standard design features was based upon the results of validation testing which was the culmination of the five-year program mentioned above. The development program and validation testing are described in Appendix 18G of the ABWR SSAR (GE Document 23A6100).

RAI Number: HHFB.4.7

Question:

SSAR Section 18.5 states that the remote shutdown system (RSS) design is described in SSAR Subsections 7.4.1.4 and 7.4.2.4, and that the controls and instrumentation required for system operation are discussed in SSAR Subsection 7.4.1.4.4. None of these subsections are included in the application, and the information is not included in the application.

GE Response:

The Remote Shutdown System design is described in SSAR Section 7.4.2. The previous references were in error and will be corrected in Amendment 1 of the SSAR (see attached).

18.5 Remote Shutdown System

The Remote Shutdown System (RSS) provides a means to safely shut down the plant from outside the main control room. It provides control of the plant systems needed to bring the plant to hot shutdown, with the subsequent capability to attain cold shutdown, in the event that the control room becomes uninhabitable.

The RSS design is described in ~~Subsections 7.4.1.4 and 7.4.2.4~~ Section 7.4.2. All of the controls and instrumentation required for RSS operation are identified in ~~Subsection 7.4.1.4.4~~ Section 7.4.2 and in Figure 21.7.4-2.

The RSS uses conventional, hardwired controls and indicators to maintain diversity from the main control room. These dedicated devices are arranged in a mimic of the interfacing systems process loops.

RAI Number: HHFB.4.8

Question:

The bases, criteria, and inventory of controls, displays, and alarms for design of the RSS control panel are not contained in the application.

GE Response:

Please see the response to RAI HHFB.4.7.

RAI Number: HHFB.4.9

Question:

The application does not identify any standard design features for the RSS control system or control panel.

GE Response:

It is true that no standard design features are identified for the Remote Shutdown System. However, the design of the RSS is subject to the same detailed implementation process as is that for the main control room MMIS.

RAI Number: HHFB.4.10

Question:

The application does not state whether the process described in Appendix 18E will be applied to the design of any portion of the RSS, including the RSS control panel.

GE Response:

The revised process to be described in the revised Appendix 18E (See response to HHFB.4.5) will be applied to the design of the Remote Shutdown System (RSS). The Human Factors Engineering (HFE) Design Acceptance Criteria (DAC) to be included in SSAR Amendment 1 will also apply to the RSS.

RAI Number: HHFB.4.11

Question:

The plant systems and controls to which the SBWR MMIS design and implementation process will be applied are not explicitly identified in the application.

GE Response:

The SBWR MMIS design and implementation process as described in Appendix 18E will be applied to the Remote Shutdown System and to all of the systems which make up the plant process man-machine interface in the main control room as determined from the task analysis.

RAI Number: HHFB.4.12

Question:

The process described in Appendix 18E reiterates the process depicted in Drawing 21.18E-1. However, the application does not:

- describe the qualifications and experience of the team that developed the process described in Appendix 18E;

- identify what standards and/or guidance were used to develop the process described in Appendix 18E;

- provide the MMIS design definition used as the basis for the MMIS design and implementation process mentioned in SSAR Section 18E.3.6;

- state the purpose for each process element;

- identify who is responsible for performance of each process element; and

- describe how the individual process elements are performed, i.e., methodology to be used and the criteria to be applied.

GE Response:

Appendix 18E will be revised in Amendment 1 to the SSAR and will address all of the above items.

RAI Number: HHFB.4.13

Question:

The application does not contain a description of the human factors engineering verification and validation program to be used throughout the SBWR MMIS design and implementation process.

GE Response:

Appendix 18E will be revised in Amendment 1 to the SSAR and will address the above item.

RAI Number: HHFB.4.14

Question:

According to the process as described in Appendix 18E, completion of process Elements 1 through 6 precedes submittal of the SSAR. No information regarding the conduct, results, or documentation of these efforts are contained in the application.

GE Response:

Appendix 18E will be revised in Amendment 1 to the SSAR and will eliminate the disucssion of earlier process elements. Refer to Appendix 18G of ABWR (GE Document 23A6100) for discussion of design and development activities.

RAI Number: HHFB.4.15

Question:

The application does not discuss the methodology used for and results of the following tasks:

- operating experience review;
- system functional requirements;
- allocation of functions;
- task analysis;
- human factors verification and validation program.

GE Response:

Appendix 18E will be revised in Amendment 1 to the SSAR and will include discussions of the plans for each of the tasks listed.

RAI Number: HHFB.4.16

Question:

Appendix 18E discusses Figure 18E-1 and Table 18E.2-1; however, neither the figure nor the table are contained in the application.

GE Response:

Appendix 18E will be revised in Amendment 1 to the SSAR to resolve the above item.

RAI Number: HICB.1

Question:

In general, the SSAR has addressed the SRP but has not addressed the substantial amount of additional criteria related to the use of digital control equipment in the I&C systems that has been addressed in the ABWR design review.

GE Response:

The criteria related to the use of digital control equipment in the I&C systems is addressed in the following SBWR SSAR sections: Section 7.1.2, Identification of Safety Criteria; Section 7.3.4, Safety System Logic and Control; Section 7.3.5 Essential Multiplexing System; and Section 7.7.7, Non-Essential Multiplexing System.

Industry codes and standards applicable to the design of the SBWR Standard Plant are listed in Table 1.9-3. More detailed criteria are established in system design specifications, procurement specifications, and installation specifications.

Where applicable, the criteria related to the use of digital control equipment is the same as that used in ABWR. Any forthcoming question of a specific nature will be responded to in a more specific manner.

RAI Number: HICB.2

Question:

There is no comparison of the SBWR design to the EPRI ALWR (Passive Plant) Man-Machine Interface System (MMIS) Requirements Document.

GE Response:

An assessment of the conformance of the contents of Chapter 18 to the EPRI Passive Plant Requirements Document has been made. This is part of an overall assessment of the SBWR Standard Safety Analysis Report against the EPRI Requirements Document that is contained in Appendix 1C.

RAI Number: HICB.3

Question:

There is no submittal for the Tier 1 design description or ITAACs. The review of the SSAR must be concurrent with the review of the ITAACs.

GE Response:

The submittal of the Tier 1 design description or ITAACs is included with the February, 1993 SSAR update.

RAI Number: HICB.4

Question:

There is no specific description of the SBWR I&C systems hardware design. The SSAR states that the environmental qualification information will be submitted on February 28, 1993. The SSAR does not address electromagnetic compatibility, fiber optic qualification, or other issues specific to the digital equipment that is described in the SSAR submittal.

GE Response:

Environmental equipment qualification is addressed in Section 3.11 and Appendix 3D (submitted in February 1993).

The interconnecting fiber optic links of the multiplexing system and Safety System Logic and Control (SSLC) are not subject to electromagnetic interference (EMI) effects. Optical fiber, being a non-electrical medium, has the inherent properties of immunity to electrical noise, such as EMI, radio frequency interference (RFI), and lightning; point-to-point electrical isolation; and the absence of conventional transmission line effects. Fiber optic multiplexing is also unaffected by the radiated noise from high-voltage conductors, by high-frequency motor control drives, and by transient switching pulses from electromagnetic contactors or other switching devices.

However, the electrical-to-optical interface at the transmitting and receiving ends must still be addressed to ensure complete immunity to EMI. The control equipment containing the electrical circuitry use standard techniques for shielding, grounding, and filtering and are mounted in grounded equipment panels provided with separate instrument ground buses. Panel location, particularly in local areas, is carefully chosen to minimize noise effects from adjacent sources. The use of fiber optic cables ensures that current-carrying ground loops will not exist between the control room and local areas.

As part of the pre-operational test program, the system will be subjected to EMI testing. EMI and RFI test measurements will be developed using the guidelines described in ANSI/IEEE C63.12-1987, American National Standard for Electromagnetic Compatibility Limits – Recommended Practice. For testing susceptibility to noise generation from portable radio transceivers, tests will be developed from ANSI/IEEE C37.90.2-1987, IEEE Trial-Use Standard, Withstand Capability of Relay Systems to Radiated Electromagnetic Interference from Transceivers. Section 5.5.3 of this standard describes tests for digital equipment using clocked logic circuits.

With the system connected, each microprocessor-based controller (one at a time) will be required to demonstrate immunity to the defined conducted and radiated tests. Units shall also comply with standard surge withstand capability tests, as follows:

(a) ANSI/IEEE C62.41 (1980), Guide for Surge Voltages in Low-Voltage AC Power Circuits

(b) ANSI/IEEE C62.45 (1987), Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Circuits

For design guidance and additional test development guidance, the following military standards shall be used:

(a) MIL-STD-461C (1987), Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference

(b) MIL-STD-462 (1967), Measurement of Electromagnetic Interference Characteristics

Because of the comprehensive nature of these documents; their applicability to ground, airborne, and shipboard equipment; and the differences in requirements for the Army, Navy, and Air Force, the use of these standards shall be limited to the susceptibility requirements and limits for class A3 equipment and subsystems (ground, fixed). Within these limits, the guidelines for Army procurements only shall be used. Tests for transmitting and receiving equipment, power generators, and special purpose military devices are not applicable.

The NRC staff is working with EG&G to develop comprehensive guidance on the type of fiber optic cable, transmitter, and receiver combinations that will provide optimum compliance with qualification requirements. The guidance will be based on the existing IEEE cable standards, such as IEEE-323 and IEEE-384, on the ANSI standards for fiber optic cables, and the results of the EG&G work.

These and other issues specific to the digital equipment described in the SSAR submittal have been previously addressed in the ABWR SSAR (GE Document 23A6100)Appendix 7A and responses to Chapter 7 questions.

RAI Number: HICB.5

Question:

There is no documentation of the SBWR software design verification and validation, configuration management control, or other aspects of software design management. The SSAR does not describe the software standards and design methods to be used.

GE Response:

The SBWR software design verification and validation, configuration management control, and other aspects of software design management are described in Section 3.3, Software Development, of the SBWR Tier 1 Design Certification Document submitted in February 1993. The Software Management Plan establishes the software standards and design methods to be used.

Software standards are listed in SBWR SSAR, Section 7.3.4.5, under Software Requirements.

RAI Number: HICB.6

Question:

There is no documentation of conformance with the TMI action items. The SSAR states that this information will be submitted on February 28, 1993.

GE Response:

Conformance with the TMI action items is documented in Appendix 1A, "Response to TMI Related Matters," and was submitted on February 28, 1993, as planned.

RAI Number: HICB.7

Question:

There is no documentation of failure modes and effects analysis (FMEA) for the I&C systems. The SSAR states that a FMEA will be submitted on February 28, 1993. However, the SSAR does not describe if the I&C systems will be specifically addressed.

GE Response:

The I & C systems will not be specifically addressed with a FMEA. The basis for this is discussed in Section 1B.1 of the February 28, 1993 SBWR SSAR submittal.

RAI Number: HICB.9

Question:

The TS have not been provided. Several issues such as bypass capability and surveillance intervals and methods described in the TS must be evaluated prior to a staff final safety evaluation report.

GE Response:

GE is working closely with the NRC staff in the development of Section 3.3, Instrumentation, of the ABWR Technical Specifications. When this activity is completed, Section 3.3, Instrumentation, of the SBWR Technical Specifications (TS) will be updated to make the SBWR approach consistent with the ABWR resolution of these issues.

RAI Number: HICB.10

Question:

The applicant has not provided a defense-in-depth study to address potential common-mode failures of I&C system equipment.

GE Response

Common-mode failures of I&C system equipment have been addressed in the SBWR design as part of the probabilistic risk assessment (PRA) in SSAR Attachment 19AB, Dependent Failures. These dependencies are modeled and quantified in compliance with the ALWR EPRI requirements methodology of the EPRI Advanced Light Water Reactor Requirements Document; Vol. III, Appendix A to Chap. 1 - Rev. 3, May, 1992.

SBWR design incorporates defense-in-depth principles through maintaining separation of control and protection functions even though sensors are shared within protection systems. In addition, the shared sensors are designed within a full four division architecture with 2-out-of-4 voting logic.

Diversity principles are incorporated at both the signal and system levels: (1) Diverse parameters are monitored to automatically initiate protective actions which are also manually controllable; (2) Multiple diverse systems are available to both shut down the reactor to cool its core.

As discussed in SECY-93-087 and SECY-91-292, the NRC staff is continuing to (1) develop regulatory guidance that could be used to assess diversity; (2) define the criteria needed to satisfy the requirements for engineering activities and design implementation; and (3) develop safety classification criteria for I&C systems in ALWR designs. As stated in SECY-93-087 the NRC staff revised the initial position proposed in the draft Commission paper dated June 25, 1992, for assessing the defenses against common-mode failures in a design.

SBWR will address this issue by extending the ABWR study performed by Lawrence Livermore National Laboratory to account for design differences of SBWR and any forthcoming regulatory guidance.

RAI Number: OTSB.0

Question:

The staff has completed its initial acceptance review of the technical specification (TS) selection criteria and content for the SBWR application, as presented in Section 16.0 of the Standard Safety Analysis Report (SSAR). Based on the staff's review, further clarification is needed for the following OTSB RAIs.

GE Response:

See GE's response for RAIs OTSB.1, OTSB.2, and OTSB.3.

RAI Number: OTSB.1

Question:

The SSAR states that in accordance with the criteria of the Commission's "Policy Statement on Technical Specification Improvements," limiting conditions for operation (LCOs) are provided. In order to verify this, a complete set of LCOs for the SBWR's passive systems is needed (as well as providing schedules for those systems designated as to be determined (TBD)).

GE Response:

Following is a list of the Technical Specifications (TS) LCOs for the SBWR's passive systems. Four systems TS contain information designated as to be determined (TBD). The ICS/PCCS Pool Level, TS 3.6.2.4 information is available at this time and the SR 3.6.2.4 has been updated to provide this data (see attached). The other three systems involve Completion Times labeled TBD. These are TS 3.1.7 Standby Liquid Control (SLC) System, TS 3.8.1 DC Sources-Operating, and TS 3.8.6 Distribution System-Operating. These Completion Times are expected to be available in Amendment 1 of the SSAR.

SBWR Passive Systems LCOs

TS 3.1.7 – Standby Liquid Control (SLC)
TS 3.4.6 – Isolation Condenser System (ICS)
TS 3.5.1 – ECCS-Operating
TS 3.5.2 – ECCS-Shutdown
TS 3.6.2.3 – Passive Containment Cooling System (PCCS)

3.6 Containment Systems

3.6.2.4 ICS/PCCS Pool

LCO 3.6.2.4 ICS/PCCS Pool shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ICS/PCCS Pool inoperable.	A.1 Restore ICS/PCCS Pool to OPERABLE status.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Initiate actions to be in MODE 4.	12 hours Immediately upon achieving MODE 3.
C. Unable to attain MODE 4 as required by Required Action B.2.	C.1 Maintain reactor coolant temperature as low as practicable by use of alternate decay heat removal methods.	As soon as practical

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify for each subcompartment of the ICS/PCCS pool that the manual isolation valves are in their locked open position.	REFUELING INTERVAL
SR 3.6.2.4.2	Verify that ICS/PCCS Pool water level is at least ~~[TBD] m ([TBD] feet)~~4.4 m (14.4 feet).	31 days
SR 3.6.2.4.3	Demonstrate that each ICS/PCCS pool subcompartment vent has unobstructed air flow path through the vent line and moisture separator to the atmosphere.	REFUELING INTERVAL

RAI Number: OTSB.2

Question:

GE needs to identify the specific differences in the proposed TS requirements from those contained in Rev. 0 of NUREGs 1433 and 1434.

GE Response:

A markup of the appropriate sections of NUREGs 1433 and 1434 will be undertaken with completion and submittal in April 1994.

Because the ABWR I&C Technical Specifications (TS) are being developed now with a final submittal on January 24, 1994, the ABWR I&C TS will be marked up to reflect the differences for the SBWR I&C TS.

RAI Number: OTSB.3

Question:

GE needs to provide specific justification for the changes to the completion times and surveillance intervals in accordance with the basis for the staff's evaluation of related topical reports.

GE Response:

Where appropriate, the bases submitted with the SBWR Technical Specifications have been established based upon the BWR Standard Technical Specifications, NUREG 1433 and 1434, Rev. 0. Where the SBWR Technical Specifications differ significantly from the BWR/4 and BWR/6 designs, the bases have been expanded to provide the justification for the selected completion times and surveillance intervals.

RAI Number: PEPB.0

Question:

In general, the staff has concluded that GE's application for FDA and SBWR design certification regarding emergency preparedness requirements contained sufficient information to establish that emergency preparedness requirements have been factored into the design bases of the SBWR, with the exception of SSAR Section 1.8. The following PEPB RAI's discusses the staff's concerns regarding SSAR Section 13.3.

GE Response:

SSAR Section 1.8 has been included in the February 28, 1993 SSAR submittal.

RAI Number: PEPB.1

Question:

Table 13.3-1 contains a summary list of SBWR design considerations pertaining to emergency planning. The table lists a Technical Support Center (TSC), Emergency Operations Facility (EOF), Operations Support Center (OSC), Emergency Operations Center (EOC), Fixed or Mobile Laboratory Facilities, Post-Accident Sampling (PASS) Capability, and Onsite Decontamination Facility (ODF). The staff agrees that the EOF and EOC are not within the scope of the SBWR design. However, it is the staff's position that the OSC and ODFs are required for SBWR design certification. The staff also notes that the reference as well as the emergency preparedness requirements given for the EOC in Table 13.3-1 are apparently incorrect. The EOC is usually a state or local government offsite facility. Please provide the basis for inclusion of this facility in Table 13.3-1.

GE Response:

Table 13.3-1 does state that the Emergency Operations Center (EOC) is not within the scope of the SBWR Standard Plant. The listing will be deleted (see attached).

Table 13.3-1 SBWR Design Considerations for Emergency Planning Requirements (Continued)

Facility	Primary Document/ Section	Emergency Planning Requirements	SBWR Design Consideration
Emergency Operations Facility (EOF)	NUREG-0696/1.3.3	The EOF is an off-site support facility for the management of overall licensee emergency response, coordination of radiological and environmental assessments, and determination of recommended public protective actions. The EOF has appropriate technical data displays and plant records to assist in the diagnosis of plant conditions to evaluate the potential or actual release of radioactive materials to the environment. A senior licensee official in the EOF organizes and manages licensee off-site resources to support the TSC and the control room operators. The OSC is an on-site assembly area separate from the control room and the TSC and shall be provided for operations support personnel to report to in an emergency. There shall be direct communications between the EOF and the TSC so that the personnel reporting to the EOF can be assigned to duties in emergency operations.	The EOF is not within the scope of the SBWR Standard Plant. It is the responsibility of the COL applicant to identify the EOF and the communication interfaces for inclusion in the detailed design of the TSC and control room. (See Subsection 13.3.3.2 for COL license information requirement.)The detailed requirements are provided in Section 4 of NUREG-0696.
~~Emergency Operations Center (EOC)~~	~~NUREG-0696/II.H.6~~	~~Each licensee shall make provision to acquire data from or for emergency access to off site monitoring equipment including geophysical phenomena and radiological monitors.~~	~~Not within the scope of the SBWR Standard Plant. However, there is no impact on SBWR design.~~

RAI Number: PEPB.2

Question:

More detailed information concerning the following facilities is needed:

- OSC - provide information on the OSC for the SBWR in sufficient detail to determine that the facility will meet the requirements of Supplement 1 to NUREG-0737 and the guidance of NUREG-0696.

- ODFs - provide sufficient information to determine that the ODFs for the SBWR will be adequate in accordance with 10 CFR Part 50, Appendix E, Section IV.E.3.

- TSC - provide information on the TSC for the SBWR in sufficient detail to determine that this facility will meet the requirements of Supplement 1 to NUREG-0737 and the guidance of NUREG-0696.

- Mobile or fixed laboratory facilities - provide information on laboratory facilities for the SBWR clarifying the role of mobile or fixed laboratory facilities in the SBWR design and the provisions made to acquire data from these facilities.

- PASS - provide sufficient information to determine that the PASS for the SBWR will meet the requirements of NUREG-0737.including the onsite counting labs and their design-basis radiation levels; location of all post-accident vital areas and their access/egress routes during accident conditions.

GE Response:

Table 13.3-1 SBWR Design Considerations for Emergency Planning Requirements has been revised to provide the additional detailed information requested (see attached).

Table 13.3-1 SBWR Design Considerations for Emergency Planning Requirements

Facility	Primary Document/ Section	Emergency Planning Requirements	SBWR Design Consideration
Technical Support Center (TSC)	NUREG-0696/1.3.1	The TSC is an on-site facility located close to the control room that shall provide plant management and technical support to the reactor operating personnel located in the control room during emergency conditions. It shall have technical data displays and plant records available to assist in the detailed analysis and diagnosis of abnormal plant conditions and any significant release of radioactivity to the environment. The TSC shall be the primary communications center for the plant during an emergency. A senior official, designated by the licensee, shall use the resources of the TSC to assist the control room operators by handling the administrative items, technical evaluation, and contact with off-site activities, relieving them of these functions. The TSC facilities may also be used for performing normal functions, such as shift technical supervisor and plant operations maintenance analysis functions, as well as for emergencies.	The SBWR Standard Plant ~~will~~ complies with all the TSC design requirements. Specifically, a TSC of sufficient size to support 26 people is located in the reactor building above the control room. Display capability in the TSC is described in Subsection 18.4.2.11. The TSC is located in a Seismic Category I structure which is environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment. The room is provided with radiological protection and monitoring equipment necessary to ensure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent, to any part of the body, identical to the main control room except that in the event of an accident requiring use of the emergency breathing air system (EBAS), the TSC would be evacuated and the TSC management function would be transferred to the control room operators as described in Section 6.4. The TSC is provided with reliable voice and data communication with the main control room and EOF and reliable voice communications with the OSC, NRC Operations Centers and state and local operations centers. Control room data communication through the emergency response data system (ERDS) with the NRC Operations Centers will also be provided as appropriate.

Table 13.3-1 SBWR Design Considerations for Emergency Planning Requirements (Continued)

Facility	Primary Document/ Section	Emergency Planning Requirements	SBWR Design Consideration
Operational Support Center (OSC)	NUREG-0696/1.3.2	The OSC is an on-site assembly area separate from the control room and TSC where licensee operations support personnel report in an emergency. There is direct communications between the OSC and the control room and between the OSC and the TSC so that the personnel reporting to the OSC can be assigned to duties in support of emergency operations.	~~The OSC is not within the scope of the SBWR Standard Plant.~~ The COL applicant is responsible for identifying the OSC and the communication interfaces for inclusion in the detailed design of the control room and TSC. (See Subsection 13.3.3.1 for COL license information requirement.) The detailed requirements are provided in Section 3 of NUREG-0696. The habitability requirements of the available OSC locations in the SBWR are not comparable to that of the control room. Thus, the COL applicant's emergency plan shall include provisions for evacuation of OSC personnel in the event of a large radioactive release. The OSC communications system shall have at least one dedicated telephone extension to the control room, one dedicated telephone extension to the TSC, and one touch-tone telephone capable of reaching on-site and off-site locations, as a minimum. Portable radio communications are also contemplated to be specified by the COL applicant.

Table 13.3-1 SBWR Design Considerations for Emergency Planning Requirements (Continued)

Facility	Primary Document/ Section	Emergency Planning Requirements	SBWR Design Consideration
Emergency Operations Facility (EOF)	NUREG-0696/1.3.3	The EOF is an off-site support facility for the management of overall licensee emergency response, coordination of radiological and environmental assessments, and determination of recommended public protective actions. The EOF has appropriate technical data displays and plant records to assist in the diagnosis of plant conditions to evaluate the potential or actual release of radioactive materials to the environment. A senior licensee official in the EOF organizes and manages licensee off-site resources to support the TSC and the control room operators. The OSC is an on-site assembly area separate from the control room and the TSC and shall be provided for operations support personnel to report to in an emergency. There shall be direct communications between the EOF and the TSC so that the personnel reporting to the EOF can be assigned to duties in emergency operations.	The EOF is not within the scope of the SBWR Standard Plant. It is the responsibility of the COL applicant to identify the EOF and the communication interfaces for inclusion in the detailed design of the TSC and control room. (See Subsection 13.3.3.2 for COL license information requirement.)The detailed requirements are provided in Section 4 of NUREG-0696.
~~Emergency Operations Center (EOC)~~	~~NUREG-0696/II.H.6~~	~~Each licensee shall make provision to acquire data from or for emergency access to off-site monitoring equipment including geophysical phenomena and radiological monitors.~~	~~Not within the scope of the SBWR Standard Plant. However, there is no impact on SBWR design.~~

Table 13.3-1 SBWR Design Considerations for Emergency Planning Requirements (Continued)

Facility	Primary Document/ Section	Emergency Planning Requirements	SBWR Design Consideration
Laboratory Facilities, Fixed or Mobile	NUREG-0654/ II.H.6.c	Provisions shall be made to acquire data from or for emergency access to off-site monitoring and analysis equipment for laboratory facilities, fixed or mobile.	The SBWR design is compatible with this requirement. Laboratory facilities for the SBWR Standard Plant will be located in the service building which is of conventional size and design. The service building is not included in the SBWR Standard Plant design scope. Provisions for emergency access to off-site monitoring and analysis equipment and laboratory facilities, fixed or mobile, will depend upon arrangements under the purview of the COL licensee. These provisions will depend upon the site location and COL licensee and support organization existing capabilities and arrangements for handling similar or duplicate activities.
Post accident sampling capability	NUREG-0737/II.B.3	Post accident sampling capability	Post Accident Sampling System of Subsection 9.3.2 meets requirements except as described in Section 1A.2.7 for the upper limit of activity in the samples at the time they are taken.
On-Site Decontamination Facility	10CFR50 Appendix E/ IV.E.3	Provisions shall be made for the decontamination of on-site individuals.	Decontamination facilities and supplies at the site for decontamination of on-site individuals will be provided by the COL applicant in the reactor building adjacent to the main change rooms. (See Subsection 13.3.3.3 for COL license information requirement.) Showers and waste collection equipment will be used to ensure spread of contamination is controlled and disposal cost of waste material is minimized. The central location is convenient to health physics support personnel who will supervise this activity.

RAI Number: PRPB.0

Question:

SSAR Chapter 12 has been partially reviewed by the staff. The general finding of the staff's review is that SSAR Chapter 12 is incomplete. Specifically, the description of in-plant airborne and contained radioactive sources are inadequate. Contained sources in the radwaste building have not been submitted. A note in SSAR Section 12.2.5 indicates that this will be submitted in February 1993. The plant locations and source geometries are not given for those contained sources that are described in Chapter 12. No in-plant airborne radioactive sources are described for the SBWR design. In addition, our review found significant omissions/deficiencies in the radiation zone diagrams provided. Some diagrams are missing. The staff counted 8 plant layout figures (Figures 21.1.2-2, Sheets 1 through 21.1.2.4) that do not have corresponding radiation zone figures. Missing features on the zone diagrams that are provided included: boundaries for the contamination/radiation control areas and their access traffic patterns; identification of very high radiation areas, as defined in 10 CFR Part 20; location of health physics (HP) facilities, including the onsite counting labs and their design-basis radiation levels; location of all post-accident vital areas and their access/egress routes during accident conditions.

GE Response:

a) Please see item (c) below for discussion of airborne contamination. Radwaste sections have been submitted in the February 28, 1993 SSAR submittal.

b) A separate table of approximate source geometries and locations with respect to the radiation zone drawings will be provided in a future SSAR amendment.

c) In-plant airborne contaimination will be evaluated in accordance with the Radiation Protection design acceptance criteria (DAC) and will be a COL applicant requirement.

d) Only floor levels and intermediate levels where radiations zones were changed were provided in the radiation zone package. The base drawings also show other intermediate levels for clarification of steel or equipment arrangement which do not change the radiation zone. A correspondence table is shown below.

e) (1) Radiation/contamination areas are clearly defined. Access traffic patterns will be added in a future SSAR amendment.

e) (2) Please provide clarification. The reactor building and radwaste building are divided into zones A through H and zones A through F in the Turbine Buildings. All drawings will be revised to the same zone designations in a future SSAR amendment.

e) (3) Health Physics facilities and counting facilities will be clearly identified in a future SSAR amendment after the service building functionality is completed. Currently, monitoring and change facilities are found in all three main buildings with a counting facility identified in the radwaste building.

e) (4) LOCA access pathways are identified but the definition of LOCA post accident areas and routes will be expanded and clarified in a future SSAR amendment.

RAI Number: RPEB.1

Question:

SSAR Section 14.2.1.1 Construction Test Objectives

SSAR Section 14.2.1.1 provides the purpose and scope of the construction and installation test program. This section states that the test abstracts will not be provided. It is the staff's position that GE should state how the construction and installation tests will be developed and who will be responsible for performing those tests.

GE Response:

The AE/Construction company will decide the details of the construction and installation test program along with the COL Applicant. For the Design Certification, it is not appropriate that the NSSS designer establishes responsibility for this testing.

RAI Number: RPEB.2

Question:

SSAR Section 14.2.2 Test Procedures

SSAR Section 14.2.2 discusses, in part, review, evaluation, and approval of initial plant test results. It is stated that the final approval of test results is obtained from the Startup Coordinating Group and the appropriate level of plant management as defined in the Startup Administrative Manual. It is the staff's position that this section of the SSAR should be modified to clarify that the review and approval of preoperational test results are normally required prior to fuel loading. If portions of any preoperational tests are intended to be conducted, or their results approved, after fuel loading, the staff has determined that the applicant referencing the GESBWR-DC should be required to: (1) list each test; (2) state which portions of each test will be delayed until after fuel loading; (3) provide technical justification for delaying these portions; and (4) state the power levels where each test will be completed.

GE Response:

GE agrees that the review and approval of preoperational test results are normally required prior to fuel loading. Some preoperational tests may be postponed until after initial fuel load and during the startup test phase. The combined operating license (COL) applicant is required to document the four requirements stated above, as described in SSAR Section 14.2.6, first paragraph.

RAI Number: RPEB.3

Question:

SSAR Section 14.2.3 Test Program's Conformance with Regulatory Guides

It is the staff's position that SSAR Section 14.2.3 should be modified to include the following items:

• RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," in accordance with SRP Section 14.2;

• The applicable portion (shutdown cooling) of RG 1.139, "Guidance for Residual Heat Removal," in accordance with SRP Section 14.2.

GE Response:

SSAR Section 14.2.3 will be modified as the staff's position indicates in Amendment 1 of the SSAR (see attached).

- Regulatory Guide 1.95 — "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," in accordance with SRP Section 14.2;
- The application portion (shutdown cooling) of RG 1.139, "Guidance for Residual Heat Removal," in accordance with SRP Section 14.2.
- Regulatory Guide 1.140 — "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants."

14.2.4 Utilization of Reactor Operating and Testing Experience in the Development of Test Program

Since every reactor/plant in a GE-NE BWR product line is an evolutionary development of the previous plant in the product line (and each product line is an evolutionary development from the previous product line), it is evident that the SBWR plants have the benefits of experience acquired with the successful and safe startup of more than 30 previous BWR-1/2/3/4/5 and BWR-6 plants. The operational experience and knowledge gained from these plants and other reactor types have been factored into the design and test specifications of GE-NE-supplied systems and equipment that will be demonstrated during the preoperational and startup test programs. Additionally, reactor operating and testing experience of similar nuclear power plants obtained from NRC Licensee Event Reports and through other industry sources will be utilized to the extent practicable in developing and carrying out the initial test program.

14.2.5 Trial Use of Plant Operating and Emergency Procedures

To the extent practicable throughout the preoperational and initial startup test program, test procedures will utilize operating, emergency, and abnormal procedures where applicable in the performance of tests. The use of these procedures is intended to do the following:

- prove the specific procedure or illustrate changes which may be required;
- provide training of plant personnel in the use of these procedures; and
- increase the level of knowledge of plant personnel on the systems being tested.

A testing procedure utilizing an operating, emergency, or abnormal procedure will reference the procedure directly, extract a series of steps from the procedure, or both, in a way that is optimum to accomplishing the above goals while efficiently performing the specified testing.

RAI Number: RPEB.4

Question:

SSAR Section 14.2.4 Utilization of Reactor Operating and Testing Experience in the Development of Test Program

While the staff agrees with the statement in SSAR Section 14.2 that many parts of the SBWR plant design have the benefits of experience acquired with the successful and safe startup of more than 30 previous BWR design plants, the SBWR design is one of the first standardized nuclear power plant designs which uses simplified, inherent, and passive means to accomplish its safety functions. Therefore, it is the staff's position that SSAR Section 14.2.4 should address a review of the vendor test program as required by 10 CFR 52.47(b)(2)(i)(A) and incorporate results of the vendor test program into the initial test program, as appropriate.

GE Response:

GE agrees that the safety features must be demonstrated by one of the methods described in 10CFR52.47(b)(2)(i)(A), but this regulation appears to apply to certification and not the initial plant test program. Once a design reaches the initial plant test phase, the Design Certification will have been issued.

GE intends to utilize the information gained in the experiences, tests and analyses to enhance the startup testing program. Additionally, the startup test program includes testing to insure that Chapter 15 analyses are valid. Vendor test program information will be utilized to meet requisition specifications and to understand SSC limitations, but it will be the operational and safety limits that will be the focus of the initial test program, not SSC ultimate capability. Vendor tests are more useful to establish component suitability for a particular application during the component selection process and directly applied in the construction and installation phase.

10 CFR 52.47(b)(2)(i)(A), April 30, 1992, addresses what must be done for Certification and does not make a specific reference to the initial test program testing or a vendor test program. Information required by 10CFR52.47(b)(2)(i)(A) has been provided by GE to the NRC letter dated May 7, 1993, and will be included in Amendment 1 of the SSAR.

RAI Number: RPEB.5

Question:

SSAR Section 14.2.7 Test Program Schedule and Sequence

It is the staff's position that SSAR Section 14.2.7 should address the requirement of the applicant referencing the GESBWR-DC to: (1) list each test, that will not be performed prior to exceeding 25-percent power, for all plant structures, systems, and components that are relied upon to prevent, to limit, or to mitigate the consequences of postulated accidents; (2) provide technical justification for delaying the tests; and (3) state the power levels where each test will be completed.

GE Response:

SSAR section 14.2.7 lists the test program schedule and sequence and states, in part, that "... To the extent practicable, the schedule should establish that, prior to exceeding 25% power, the test requirements will be met for those plant structures, systems, and components that are relied on to prevent, limit, or mitigate the consequences of postulated accidents." This means that the testing shall be accomplished as scheduled in the preoperational testing phase and as required by Table 14.2-1. If not, the COL Applicant should justify it accordingly. This is worded just as the equivalent section of the ABWR SSAR and appears to address the question.

RAI Number: RPEB.7

Question:

General Comments on SSAR Section 14.2

It is the staff's position that individual tests listed in Section 14.2 of the SSAR for structures, systems, components, and features that are not essential to the demonstration of conformance with design requirements important to safety, but which meet any of the following criteria, should be identified.

• Those that will be used for shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period.

• Those that will be used for shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.

• Those that will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications.

• Those that are classified as engineered safety features or will be used to support or ensure the operations of engineered safety features within design limits.

• Those that are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the SSAR.

• Those that will be used to process, store, control, or limit the release of radioactive materials.

GE Response:

This question asks for lists of tests for structures, systems, components and features that are not essential to the demonstration of conformance with design requirements important to safety. Systems that are both safety-related and non-safety-related and meet the criteria are provided in the following lists in the order of the six criteria:

1) The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System is used for cold shutdown of the reactor during normal plant conditions and for an extended shutdown period. It is a non-safety-related system. The applicable sections are 14.2.8.1.6 and 14.2.8.2.18.

2) Systems used to reach the safe shutdown condition during transient conditions are the engineered safety features listed in the response to item 4). The RWCU/SDC System listed in response 1) is the non-safety-related system used to reach cold shutdown during

infrequent or moderately frequent events. Tests for infrequent or moderately frequent events are listed in Subsections 14.2.8.2.24 and 14.2.8.2.28.

3) There are no preoperational or startup tests that establish conformance with Safety Limits or Limiting Conditions for Operation. These limits are controlled by ITACC tests and by operating procedures developed by the COL applicant.

4) The engineered safety features are listed in Section 6.0. The systems are the Containment, Containment Isolation System, the Safety System Logic and Control (SSLC), the Leak Detection and Isolation System (LDIS), the Isolation Condenser (IC) System, the Passive Containment Cooling System (PCCS), the safety envelope, the Flammability Control System (FCS), the Gravity-Driven Cooling System (GDCS), and the Automatic Depressurization Subsystem (ADS), the Sealed Emergency Operating Area (SEOA), and the Emergency Breathing Air System (EBAS). They are all safety-related systems. Applicable sections are 14.2.8.1.1, 14.2.8.1.7, 14.2.8.1.9, 14.2.8.22, 14.2.8.2.23, 14.2.8.2.29, 14.2.8.1.30, 14.2.8.1.31,14.2.8.1.32, 14.2.8.1.33, 14.2.8.1.34, 14.2.8.1.35, 14.2.8.1.36, 14.2.8.1.37, 14.2.8.1.40, 14.2.8.1.67, 14.2.8.1.70, 14.2.8.1.71, 14.2.8.1.72, 14.2.8.1.73, and 14.2.8.1.74.

5) These systems are the engineered safety features listed to in response 4).

6) The Gaseous Radwaste System, the Solid Radwaste System, and the Liquid Radwaste System are described in Chapter 11 of the SSAR. These are non-safety-related systems. Applicable sections are 14.2.8.2.30, 14.2.8.2.53, 14.2.8.1.45, 14.2.8.1.53, and 14.2.8.1.69.

RAI Number: RPEB.8

Question:

General Comments on SSAR Section 14.2.8

It is the staff's position that the Table of Contents in Sections 14.2.8.1 and 14.2.8.2 should be extended to list all preoperational tests and startup tests covered in Section 14.2 of the SSAR.

GE Response:

GE concurs and will revise the Table of Contents for Chapter 14 in Amendment 1 to include two levels of entries below Subsection 14.2.8.

RAI Number: RPEB.9

Question:

Chapter 17.3, Reliability Assurance Program During Design Phase, GE initially developed a design reliability assurance program (D-RAP) for the ABWR. Most of the specific comments listed in the following RPEB RAIs are based on the differences the staff noted between the text of the ABWR and SBWR D-RAP submittals. GE should identify if these two programs are to be maintained independently or if common methodology and management of the programs will be used as, for example, in the Quality Assurance Program.

SSAR Section 17.3.1 states that a plant owner/operator will have an operational reliability assurance program (O-RAP). However, an owner/operator will also be required to have a D-RAP for those risk-significant systems, structures, and components (SSCs) that are not covered by the GE-NE D-RAP and those risk-significant SSCs that are designed or procured by the owner/operator or their agent. GE should clarify that SSAR Section 17.3.1 describes the GE-NE D-RAP and GE should state that an owner/operator will be required to provide both a D-RAP and an O-RAP.

GE Response:

The text of both the ABWR and the SBWR Reliability Assurance Programs should be identical for both programs with the exception that the examples will be different for ABWR and SBWR. The text of the SBWR will be revised in Amendment 1 (see attached) to state that the owner/operator will be required to provide a D-RAP for those risk-significant structures, systems, and components (SSCs), if any, that are not covered by the GE-NE D-RAP, and an O-RAP. (The attached draft SSAR Amendment 1 also applies to attachments mentioned in RAIs RPEB.11, RPEB.14, RPEB.15, and RPEB.16.)

17.3 Reliability Assurance Program During Design Phase

This section presents the SBWR Design Reliability Assurance Program (D-RAP).

17.3.1 Introduction

The SBWR Design Reliability Assurance Program (D-RAP) is a program that will be performed ~~by GE Nuclear Energy (GE-NE)~~ during detailed design and specific equipment selection phases to assure that the important SBWR reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout the plant life. The plant owner/operator will complete the D-RAP for those risk-significant structures, systems, and components, if any, that are not covered by the GE-NE D-RAP and will also have an Operational Reliability Assurance Program (O-RAP) that tracks equipment reliability to demonstrate that the plant is being operated and maintained consistent with PRA assumptions so that overall risk is not unknowingly degraded.

The PRA evaluates the plant response to initiating events to assure that plant damage has a very low probability and risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of the plant risk-significant structures, systems and components (SSCs) throughout plant life. SSAR Appendix 19K, PRA Based Reliability and Maintenance, identifies certain risk-significant SSCs. The results of Appendix 19K can be used as a starting point for the D-RAP.

The D-RAP will include the design evaluation of the SBWR. It will identify relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for owner/operator consideration in assuring safety of the equipment and limited risk to the public. The policy and implementation procedures will be specified by the owner/operator.

Also included in this explanation of the D-RAP is a descriptive example of how the D-RAP will apply to one potentially important plant system, the Isolation Condenser System (ICS). The ICS example shows how the principles of D-RAP will be applied to other systems identified by the PRA as being significant with respect to risk.

17.3.2 Scope

The SBWR D-RAP will include the future design evaluation of the SBWR, and it will identify relevant aspects of plant operation, maintenance, and performance monitoring of plant risk-significant SSCs. The PRA for the SBWR and other industry sources will be used to identify and prioritize those SSCs that are important to prevent or mitigate plant transients or other events that could present a risk to the public.

17.3.3 Purpose

The purpose of the D-RAP is to assure that the plant safety as estimated by the PRA is maintained as the detailed design evolves through the implementation and procurement phases and that pertinent information is provided in the design documentation to the future owner/operator so that equipment reliability, as it affects plant safety, can be maintained through operation and maintenance during the entire plant life.

17.3.4 Objective

The objective of the D-RAP is to identify those plant SSCs that are significant contributors to risk, as shown by the PRA or other sources, and to assure that, during the implementation phase, the plant design continues to utilize risk-significant SSCs whose reliability is commensurate with the PRA assumptions. The D-RAP will also identify key assumptions regarding any operation, maintenance and monitoring activities that the owner/operator should consider in developing its O-RAP to assure that such SSCs can be expected to operate throughout plant life with reliability consistent with that assumed in the PRA.

A major factor in plant reliability assurance is risk-focused maintenance, by which maintenance resources are focused on those SSCs that enable the SBWR safety-related systems to fulfill their safety-related functions and on SSCs whose failure may directly initiate challenges to safety-related systems. All plant modes are considered, including equipment directly relied upon in emergency operating procedures (EOPs). Such a focus of maintenance will help to maintain an acceptably low level of risk, consistent with the PRA.

17.3.5 GE-NE Organization for D-RAP

~~The relevant portion of a typical GE-NE organization chart for a SBWR D-RAP is shown in Figure 17.3-1. The Manager of the Nuclear Services and Projects Department reports to the Vice President and General Manager of GE-NE Nuclear Energy. Two sections involved with an SBWR D-RAP are the Advanced Reactor Programs Section and the Engineering Services Section.~~

~~Authority for the management of an SBWR program is centered with the Advanced Reactor Programs Manager. Day to day details of an SBWR program are directed by the Project Manager, who reports to the Advanced Reactor Programs Manager. The Project Manager and his staff coordinate both the GE-NE support for the Project and the work of external organizations, such as the architect engineer.~~

The D-RAP definition, reliability analyses, and the PRA, including Appendix 19K, were performed by GE Nuclear Energy (GE-NE).

Responsibility for the design of key equipment, components and subsystems ~~is~~ was shared by GE-NE ~~the several units in the Advanced Reactor Programs Section~~ together with external organizations, including the architect engineer. ~~Reporting directly to each engineering functional manager will be performing engineers, including system designers and component designers. Design support will also be provided by other design sections within GE-NE and the Nuclear Services and Projects Department. Responsibility for SBWR safety analysis and PRA studies is under the Systems Integration and Performance Engineering Unit.~~

The manager ~~Manager, System Integration and Performance Engineering, will be~~ assigned the responsibility of managing and integrating the D-RAP Program. ~~He will have~~ had direct access to the SBWR Project Manager and kept ~~will keep~~ him abreast of D-RAP critical items, program needs and status. He had ~~has~~ organizational freedom to:

- Identify D-RAP problems.
- Initiate, recommend or provide solutions to problems through designated organizations.
- Verify implementation of solutions.
- Function as an integral part of the final design process.

~~Reliability analyses, including the PRA, are performed by the Reliability Engineering Services Unit in the Licensing and Consulting Services Subsection of the Engineering Services Section (Figure 17.3-1). Thus, the PRA input to the D-RAP and many of the SBWR reliability analyses will be performed in this organization, within the Nuclear Services and Projects Department. Responsibility for reliability review of designed SBWR systems and components also falls on the Reliability Engineering Services Unit, under direction from the Systems Integration and Performance Engineering Unit.~~

The combined operating license applicant will need to supply a D-RAP organization description at the time of application for these risk-significant SSCs that are designed or procured by the applicant.

17.3.6 SSC Identification/Prioritization

The PRA prepared for the SBWR will be the primary source for identifying risk-significant SSCs that should be given special consideration during the detailed design and procurement phases and/or considered for inclusion in the O-RAP. The method by which the PRA is used to identify risk-significant SSCs is described in Chapter 19. It is also possible that some risk-significant SSCs will be identified from sources other than the PRA, such as nuclear plant operating experience, other industrial experience, and relevant component failure data bases.

17.3.7 Design Considerations

The reliability of risk-significant SSCs, which are identified by the PRA and other sources, will be evaluated at the detailed design stage (under contract to the combined operating license applicant) by appropriate design reviews and reliability analyses. Current data bases will be used to identify appropriate values for failure rates of equipment as designed, and these failure rates will be compared with those used in the PRA. Normally the failure rates will be similar, but in some cases they may differ because of recent design or data base changes. Whenever failure rates of designed risk-significant SSCs are significantly greater than those assumed in the PRA, an evaluation will be performed to determine if the equipment is acceptable or if it must be redesigned to achieve a lower failure rate.

For those risk-significant SSCs, as indicated by PRA or other sources, component redesign (including selection of a different component) will be considered as a way to reduce the core damage frequency (CDF) contribution. (If the system unavailability or the CDF is acceptably low, less effort will be expended toward redesign.) If there are practical ways to redesign a risk-significant SSC, it will be redesigned and the change in system fault tree results will be calculated. Following the redesign phase, dominant SSC failure modes will be identified so that protection against such failure modes can be accomplished by appropriate activities during plant life. The design considerations that will go into determining an acceptable, reliable design and the SSCs that must be considered for O-RAP activities are illustrated in ~~Figure 17.3-2~~ Figure 17.3-1.

GE-NE will identify in the PRA or other design documents to the plant owner/operator the risk-significant SSCs and their associated failure modes and reliability assumptions, including any pertinent bases and uncertainties considered in the PRA. GE-NE will also provide this information for the plant owner/operator to incorporate into the O-RAP to help assure that PRA results will be achieved over the life of the plant. This information can be used by the owner/operator for establishing appropriate reliability targets and the associated maintenance practices for achieving them.

17.3.8 Defining Failure Modes

The determination of dominant failure modes of risk-significant SSCs will include historical information, analytical models and existing requirements. Many BWR systems and components have compiled a significant historical record, so an evaluation of that record comprises Assessment Path A in ~~Figure 17.3-3~~ Figure 17.3-2. Details of Path A are shown in ~~Figure 17.3-4~~ Figure 17.3-3.

For those SSCs for which there is not an adequate historical basis to identify critical failure modes, an analytical approach is necessary, shown as Assessment Path B in ~~Figure 17.3-3~~ Figure 17.3-2. The details of Path B are given in ~~Figure 17.3-5~~ Figure 17.3-4. The failure modes identified in Paths A and B are then reviewed with respect to the existing maintenance activities in the industry and the maintenance requirements, Assessment Path C in ~~Figure 17.3-3~~ Figure 17.3-2. Detailed steps in Path C are outlined in ~~Figure 17.3-6~~ Figure 17.3-5.

17.3.9 Operational Reliability Assurance Activities

Once the dominant failure modes are determined for risk-significant SSCs, an assessment is required to determine suggested O-RAP activities that will assure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, and/or periodic preventive maintenance (Reference 17.3-1). An example of a decision tree that would be applicable to these activities is shown in ~~Figure 17.3-7~~ Figure 17.3-6. As indicated, some SSCs may require a combination of activities to assure that their performance is consistent with that assumed in the PRA.

Periodic testing of SSCs may include startup of standby systems, surveillance testing of instrument circuits to assure that they will respond to appropriate signals, and inspection of SSCs (such as tanks and pipes) to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurement of output (such as pump flow rate or heat exchanger temperatures), measurement of magnitude of an important variable (such as vibration or temperature), and testing for abnormal conditions (such as oil degradation or local hot spots).

Periodic preventive maintenance is an activity performed at regular intervals to preclude problems that could occur before the next preventive maintenance (PM) interval. This could be regular oil changes, replacement of seals and gaskets, or refurbishment of equipment subject to wear or age related degradation.

Planned maintenance activities will be integrated with the regular operating plans so that they do not disrupt normal operation. Maintenance that will be performed more frequently than refueling outages must be planned so as to not disrupt operation or be likely to cause reactor scram, engineered safety feature (ESF) actuation, or abnormal transients. Maintenance planned for performance during refueling outages must be conducted in such a way that it will have little or no impact on plant safety, on outage length or on other maintenance work.

17.3.10 Owner/Operator's Reliability Assurance Program

The O-RAP that will be prepared and implemented by the SBWR owner/operator will make use of the information provided by GE-NE. This information will help the owner/operator determine activities that should be included in the O-RAP. Examples of elements that might be included in an O-RAP are as follows:

Reliability Performance Monitoring — Measurement of the performance of equipment to determine that it is accomplishing its goals and/or that it will continue to operate with low probability of failure.

Reliability Methodology — Methods by which the plant owner/operator can compare plant data to the SSC data in the PRA.

Problem Prioritization — Identification, for each of the risk-significant SSCs, of the importance of that item as a contributor to its system unavailability and assignment of priorities to problems that are detected with such equipment.

Root Cause Analysis — Determination, for problems that occur regarding reliability of risk-significant SSCs, of the root causes, those causes which, after correction, will not recur to again degrade the reliability of equipment.

Corrective Action Determination — Identification of corrective actions needed to restore equipment to its required functional capability and reliability, based on the results of problem identification and root cause analysis.

Corrective Action Implementation — Carrying out identified corrective action on risk-significant equipment to restore equipment to its intended function in such a way that plant safety is not compromised during work.

Corrective Action Verification — Post-corrective action tasks to be followed after maintenance on risk-significant equipment to assure that such equipment will perform its intended functions.

Plant Aging — Some of the risk-significant equipment is expected to undergo age related degradation that will require equipment replacement or refurbishment.

Feedback to Designer — The plant owner/operator will periodically compare performance of risk-significant equipment to that specified in the PRA and D-RAP, and, at its discretion, may send SSC performance data to plant or equipment designers in those cases that consistently show performance below that specified.

Programmatic Interfaces — Reliability assurance interfaces related to the work of the several organizations and personnel groups working on risk-significant SSCs.

The plant owner/operator's O-RAP will address the interfaces with construction, startup testing, operations, maintenance, engineering, safety, licensing, quality assurance and procurement of initial and replacement equipment.

17.3.11 D-RAP Implementation

An example of implementation of the D-RAP is given for the ICS. For this example it is assumed that some ICS components have been identified by the PRA as making a significant contribution to the core damage frequency (CDF).

The purpose of the ICS is to control reactor pressure and water level within acceptable ranges so that emergency reactor depressurization trips will not occur following reactor isolation and shutdown from full power without feedwater makeup. The ICS must also, over a longer duration, remove excess sensible and core decay heat from the reactor with minimal loss of coolant inventory from the reactor when the normal heat removal systems are unavailable for any reason.

The ICS is expected to operate during transients for a reactor pressure vessel (RPV) gauge pressure between 6.205 and 8.618 MPa (900-1250 psig).

17.3.11.1 ICS Description

The ICS basically consists of three high pressure, totally independent loops, each containing a condenser that condenses steam on the tube side and transfers heat to water in a large pool, the isolation condenser/passive containment cooling (IC/PCC) pool, which is positioned above and outside the containment (drywell). The surface of the pool is vented to the atmosphere. A simplified ICS P&ID is shown in ~~Figure 17.3-8~~ Figure 17.3-7. (Refer to Section 5.4.6 for a detailed ICS System description which is summarized below.)

The condenser is connected by piping to the RPV and is placed at an elevation above the source of steam. When the steam is condensed, it returns to the vessel through a condensate return pipe. The steam side connection between the vessel and the IC is normally–open and the condensate line is normally–closed. This allows the isolation condenser and drain piping to fill with condensate which is maintained at a subcooled temperature by the pool water during normal reactor operation.

The steam supply line is vertical and feeds two horizontal headers through four pipes. The steam line is properly insulated and enclosed in a guard pipe which penetrates the containment roof slab. Two normally-open, fail-as-is isolation valves in series (nitrogen-motor-operated F001 and motor-operated F002) are located in the run of steam supply line piping inboard of the containment boundary. They are used to isolate that part of the ICS that is located outside the containment. Two different valve actuator types are used to provide diverse means for flow path closure.

Steam is condensed inside vertical tubes of the condenser and is collected in two lower headers. Two pipes, one from each lower header, take the condensate to the common drain line which vertically penetrates the containment roof slab. On the condensate return piping, two normally-open, fail-as-is isolation valves in series (motor-operated than 6.7 meters (264 in.)to provide adequate column height for natural circulation flow.

The pool subcompartment interconnections are as follows: The individual IC/PCC pool subcompartments are connected to the other pool subcompartments below the water level by locked-open valves, one for each subcompartment, so that each IC has access to the entire pool. These valves can be closed to isolate and empty the individual partitioned IC pool for maintenance of the unit. All other pool subcompartments are interconnected below the pool water level.

The water volume above the top of the IC tubes is at least 1100 cubic meters (38,846 cubic feet) in order to meet the 72 hours decay heat boiloff requirement. The remote handwheels on the locked open valves extend above the water level to locations accessible to the operator.

The walls containing the airspace flow path extend above the normal water level; this enhances the flow stability and heat removal capability of the condensers by establishing a flow path for steam leaving the pool and for the pool make-up water through the lower pipes.

17.3.11.2 ICS Operation

During normal plant operation, the IC loops are in "ready standby," so ICS operation will start upon opening of one valve. Both steam supply isolation valves and both isolation valves on the condensate return line are in a normally-open position, the condensate level in the IC extends above the upper headers, the condensate return valves are both closed, and the small vent lines from the IC top and bottom headers to the suppression pool are closed.

A small amount of steam flows from the steam piping above the ICs through the purge line by the pressure differential caused by main steam line flow. For each IC loop the four normally–open~~-open, nitrogen operated~~ isolation valves (two nitrogen-operated gate valves and two motor-operated gate valves) fail as is; the four normally–closed, solenoid-operated vent valves (globe valves) fail closed; the two normally–closed, motor-operated vent valves (globe valves) fail as is; the normally–closed, motor-operated condensate return valve (gate valve) fails as is; the normally–closed, nitrogen-operated condensate return bypass valve (globe valve) fails open; and the normally–open, motor-operated purge line valve (globe valve) fails as is.

During refueling, the IC is isolated from the reactor. All isolation valves (F001 through F004) and all vent valves (F007 through F012) are closed.

During plant operation, one of the ICS initiation signals opens the condensate return valve F005 within 30 seconds, thus starting the IC operation. If the IC does not operate, the RPV gauge pressure will increase to the SRV setpoint 8.618 MPa (1250 psig). Also, isolation valves (F001, 2, 3, and 4) are signaled to open to assure that they were reopened during or after a test closure of the valves. Condensate bypass valve F006 will open to initiate ICS operation by remote manual operation or if there is a loss of nitrogen pressure or of dc power.

If, during IC operation and after the initial transient, the RPV gauge pressure increases above 7.653 MPa (1110 psig), the bottom vent valves F009 and F010 automatically open to vent to the suppression pool. When the RPV gauge pressure decreases below ~~7.653 MPa~~7.585 MPa (1100 psig) reset value, and after a time delay to avoid too many cycles, these two valves close.

The three initiation signals which actuate all three ICS loops at the same time, opening the condensate return valve F005, are described as follows:

- The "reactor mode switch is in RUN" and the inboard or outboard MSIV position is < 90% open on both MSL(A) and MSL(B). (MSIV closure is initiated on reactor water level below L2 and other isolation closure signals). There are two main steam isolation valves (MSIVs) on each main steam line. The logic is: one-out-of-two limit switches of the MSIVs on one line plus one-out-of-two limit switches of the MSIVs on the other line (logic one-out-of-two twice). During MSIV testing, one MSL is out of service; if a one-out-of-two signal comes from the limit switches of the MSIVs of the other line, the IC goes into operation.

- RPV gauge pressure (with logic two-out-of-four) is ≥ 7.446 MPa (1080 psig) for 10 seconds or more.

- Operator manual initiation.

  When the RPV gauge pressure decreases below the IC System reset value 5.516 MPa (800 psig), the operator may stop the ICS loops individually, overriding the system initiation signals coming from closure of the MSIVs.

Condensate return valve F005 fails as is on loss of electrical power supply. Condensate return bypass valve F006 opens automatically upon a loss of the nitrogen supply, loss of two electrical power divisions, manually, by operator action, or on reactor water level below Level 2.

Automatic actuation for the vent valves (F009 and F010, located in series) is provided by a high RPV pressure (above system actuation value) and either of the condensate return valves not fully closed (with time delay to avoid the vents opening during the initial transient). The valves close, preventing loss of inventory, when the RPV pressure decreases below a reset value.

Four radiation sensors are installed in the IC/PCC pool exhaust passages that vent air and coolant vapor to the environment. Detection of a low-level leak (radiation level above background, logic two-out-of-four) initiates an alarm. Detection of a high radiation level (exceeding site boundary limits, logic two-out-of-four) isolates the leaking isolation condenser automatically (closure of isolation valves F001 through F004). The high radiation may be caused by a leak from any IC tube and a subsequent release of noble gas to the air above the IC/PCC pool.

Four redundant sets of differential pressure instrumentation (dPT) on the steam line and another four sets on the condensate return line are used to detect a possible loss-of-coolant accident (LOCA). A high dPT signal coming from two-out-of-four dPTs on the same line (steam or condensate) will result in alarms to the operator and automatic closure of all isolation valves, rendering the IC inoperable.

Alarm and closure of the isolation valves (F001 through F004) are automatic on the following signals coming from a single loop (logic two-out-of-four):

- high mass flow in the IC steam supply line;
- high mass flow in the IC condensate return line; and
- high radiation in the pool steam flow path.

The operator cannot override the high radiation signals from the IC atmosphere vents and high differential pressure IC-isolation signals.

A temperature element is provided downstream of the valves in each vent line to confirm functioning of vent valves. A temperature element is similarly provided in the condensate return line, downstream of the isolation valve F004.

17.3.11.3 Major Differences from Operating Boiling Water Reactors

The ICS design is similar to that of the few operating boiling water reactors ~~BWRs~~ (BWRs) that have ICs. Automatic and manual actuation of the SBWR ICS is similar to that incorporated in operating BWRs. The major differences for the SBWR are (1) use of three heat exchangers (HXs) instead of the one or two in operating plants; (2) use of vertical tube HXs instead of horizontal tubes; (3) use of both NOVs and MOVs for condensate return valves instead of only MOVs; and (4) use of a large pool instead of an HX shell.

The number of HXs for the SBWR is partly determined by the desire for equipment redundancy and for limiting the length and number of tubes in each HX. Vertical tube HXs of the SBWR provide for greater stability of flow and less problems with noncondensable gases. Since the condensate return bypass valves are operated by nitrogen, and fail open on loss of nitrogen pressure or electrical power, they do not require electrical power as do the motor-operated condensate return valves.

The large IC/PCC pool provides cooling water capacity for 72 hours following SBWR scram. Following that time makeup water can be provided by water trucks through safety-related piping providing makeup connections at grade level outside the reactor building. Operating BWRs have typically 20 to 30 minutes of water capacity in IC HXs, with make up provided by pumping from the condensate storage tanks or from the fire main.

17.3.11.4 Identification of Risk-Significant SSCs

An example top level fault tree for the ICS is shown in ~~Figure 17.3-9~~ Figure 17.3-8, with the top gate defined as failure of the ICS to inject water into the RPV when required. Four major events were analyzed, loss-of-coolant accidents (LOCAs), transients, loss of off-site power (LOSP) and anticipated transient without scram (ATWS). For the LOCA adequate ICS water injection is accomplished with one of the three ICs, so all three ICs must fail to result in system failure. The other events can be accommodated by any two ICs, so failure of two-out-of-three ICs results in system failure. One detail not shown in the fault tree is that, for water injection following LOCA or ATWS events, at least one vent path to the SP must be established. This means that valves F009 & F010 or F011 & F012 must open, as can be seen from ~~Figure 17.3-8~~ Figure 17.3-7.

Based upon the fault tree analysis a ranking of the ICS components or events by importance allows identification of those SSCs with greatest importance. Such components and events are shown in Table 17.3-1.

For this example, the most risk significant SSCs are listed in Table 17.3-2. These SSCs should be considered as risk-significant candidates for O-RAP activities. No SSCs appear to be risk-significant because of aging or common cause considerations.

17.3.11.5 System Design Response

The ~~three~~ two types of ICS risk-significant components identified in Table 17.3-2 as having high importance in the ICS fault tree are now considered for redesign or for O-RAP activities. The flow chart of ~~Figure 17.3-2~~ Figure 17.3-1 guides the designer.

The components identified in Table 17.3-2 are IC loop isolation valves~~, IC loop vent valves~~ and condensate return valves. The most significant failure of these valves is mechanical failure. Isolation valves have a relatively high probability of mechanical failure to open following a closure test, which is assumed to occur quarterly. Any one of the four isolation valves in each loop could disable that loop if it failed to open. Failure of a condensate return valve to open when IC operation is signaled, coupled with failure of the bypass return valve, would also disable that loop. ~~Failure of vent valves to open during certain events could leave the ICs ineffective because of noncondensable gases in the loops.~~ These ~~three~~ two components are identified for special attention with regard to reducing the risk of system failure.

Identification of Maintenance Requirements

For each identified failure mode the appropriate maintenance tasks will be identified to assure that the failure mode will be (1) avoided, (2) rendered insignificant, or (3) kept to an acceptably low probability. The type of maintenance and the maintenance frequencies are both important aspects of assuring that the equipment failure rate will be consistent with that assumed for the PRA. As indicated in Figure 17.3-7, the designer would consider periodic testing, performance testing or periodic preventive maintenance as possible O-RAP activities to keep failure rates acceptable.

For the ICS isolation valves~~, condensate return valves and vent valves~~ and condensate return valves, which normally have no required cycles during operation, a quarterly full-stroke test is judged (for this example) to be appropriate. Such tests are in compliance with ASME Code requirements for valves in nuclear plants. Detailed disassembly, inspection and refurbishment of valves would be done less frequently. Examples of maintenance activities and frequencies are shown in Table 17.3-3 for each identified failure mode. The D-RAP will include documentation of the basis for each suggested O-RAP activity.

17.3.12 Glossary of Terms

Core Damage Frequency — As calculated by the probabilistic risk assessment.

Design Reliability Assurance Program — Performed by the plant designer to assure that the plant is designed so that it can be operated and maintained in such a way that the reliability assumptions of the probabilistic risk assessment apply throughout plant life.

GE Nuclear Energy — SBWR plant designer.

Owner/Operator — The utility or other organization that owns and operates the SBWR following construction.

Operational Reliability Assurance Program — Performed by the plant owner/operator to assure that the plant is operated and maintained safely and in such a way that the reliability assumptions of the PRA apply throughout plant life.

Piecepart — A portion of a (risk-significant) component whose failure would cause the failure of the component as a whole. The precise definition of a "piecepart" will vary between component types, depending upon their complexity.

Probabilistic Risk Assessment — Performed to identify and quantify the risk associated with the SBWR.

Risk-Significant — Those structures, systems and components which are identified as contributing significantly to the ~~system unavailability~~ core damage frequency.

Table 17.3-1
ICS Components with Largest Contribution to Core Damage Frequency

Component		Fussel-Vesely Importance	Risk Achievement Worth
ICA-UNVL	IC "A" unavailable due mainly to the failure to reopen isolation valves after test	0.21	1.1
~~ICBMV005GO~~	~~Motor operated valve F005B fails to open~~		
~~ICCMV005GO~~	~~Motor operated valve F005C fails to open~~		
ICBMOD02	IC "B" mechanical failure of valve F006B	0.10	20
ICCMOD02	IC "C" mechanical failure of valve F006C	0.10	20
ICBMV005GO	Motor operated valve F005B fails to open	0.002	1.2
ICCMV005GO	Motor operated valve F005C fails to open	0.002	1.2
~~ICBKV009FA~~	~~IC "B" mechanical failure of valve F009B~~		
~~ICBKV010FA~~	~~IC "B" mechanical failure of valve F010B~~		
~~ICCKV009FA~~	~~IC "C" mechanical failure of valve F009C~~		
~~ICCKV010FA~~	~~IC "C" mechanical failure of valve F010C~~		

NOTE:
Although the "failure to reopen isolation valves after test" is assigned to IC "A", and mechanical failure of condensate return valves or IC vent valves is assigned to ICs "B" and "C", each type of failure could occur in any of the three loops.

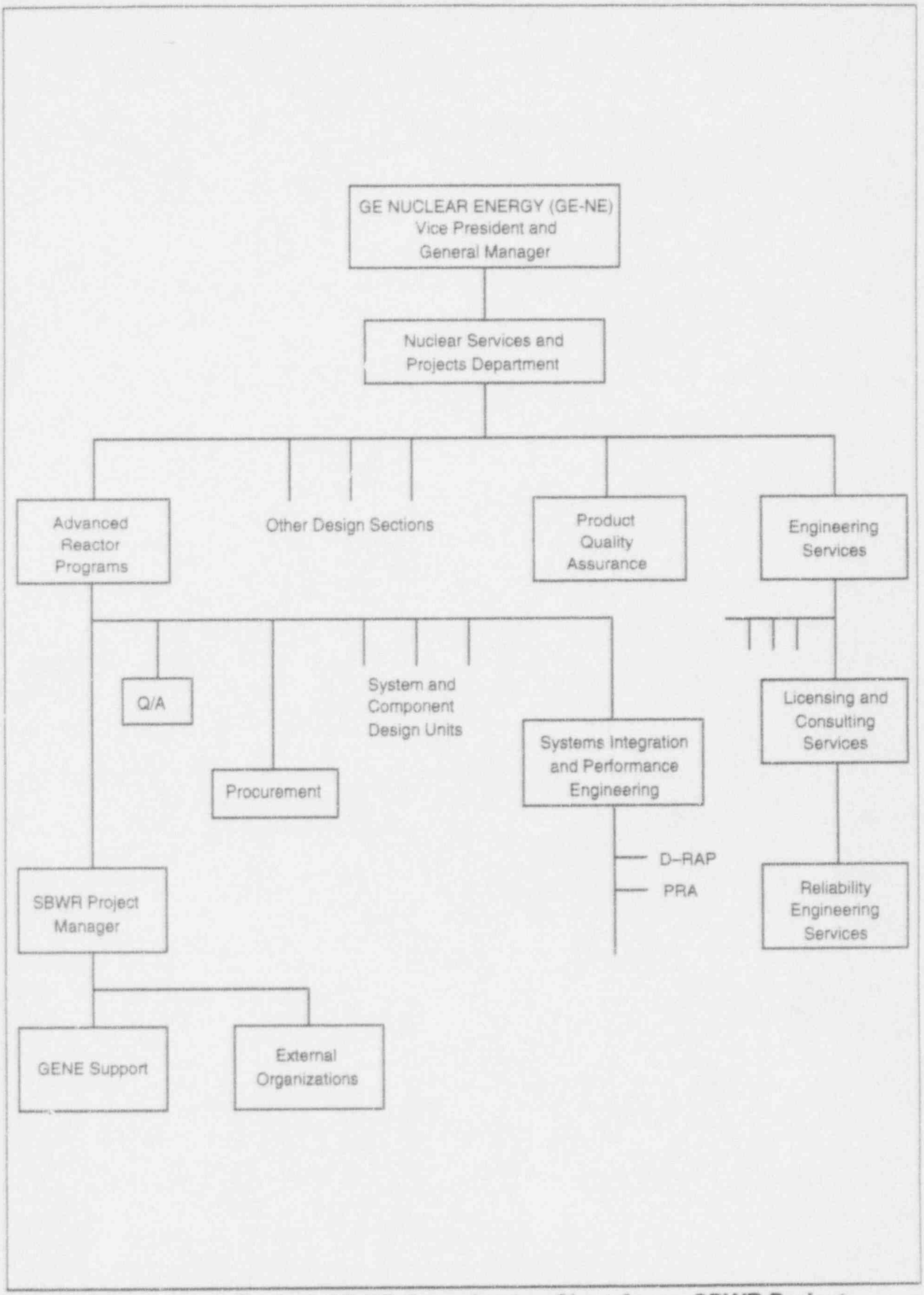
Table 17.3-2 Risk-Significant SSCs for ICS

Component	Valve Number
Isolation Valves	F001A, B & C F002A, B & C F003A, B & C F004A, B & C
Condensate Return Valves	F005A, B & C F006A, B & C
~~Vent Valves~~	~~F009A, B & C~~ ~~F010A, B & C~~

Table 17.3-3 Examples of ICS Failure Modes & O-RAP Activities

Component	Failure Mode/Cause	Recommended Maintenance	Maintenance Interval	Basis*
Isolation valves	Failure to open because of mechanical or pneumatic problems	Stroke test	3 months	Experience; ASME Code ISI.
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.
	Failure to open because of electrical problems	Electrical circuit test (may be part of stroke test)	3 months	Experience
Condensate return valves	Failure to open because of mechanical or pneumatic problems	Stroke test	3 months	Experience; ASME Code ISI.
		Visual and penetrant inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.
	Failure to open because of electrical problems	Electrical circuit test (may be part of stroke test)	3 months	Experience
~~Vent valves~~	~~Failure to open because of mechanical problems~~	~~Stroke test~~	~~3 months~~	~~Experience; ASME Code ISI.~~
		~~Visual and penetrant inspection of stem, replace if necessary.~~	~~10 years~~	~~Low failure rate; ASME Code ISI.~~
	~~Failure to open because of electrical problems~~	~~Electrical circuit test (may be part of stroke test)~~	~~3 months~~	~~Experience~~

* These types of ICS valves have been used in operating BWRs, so there is much experience to guide owners/operators in care of the equipment.



~~Figure 17.3-1 Typical GE NE Organization Chart for an SBWR Project~~

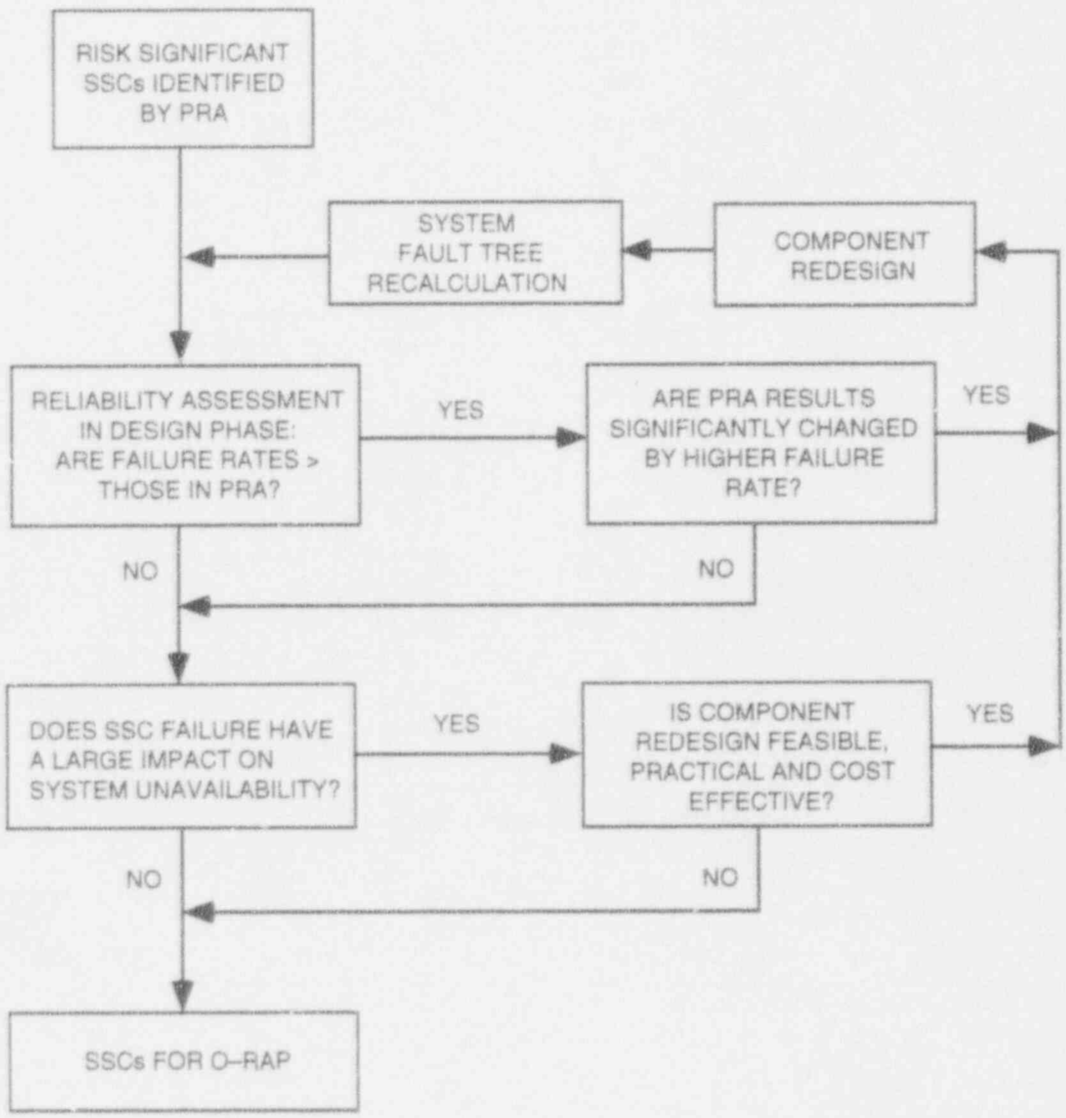


Figure 17.3-1 Design Evaluation for SSCs

~~**Figure 17.3-2 Design Evaluation for SSCs**~~

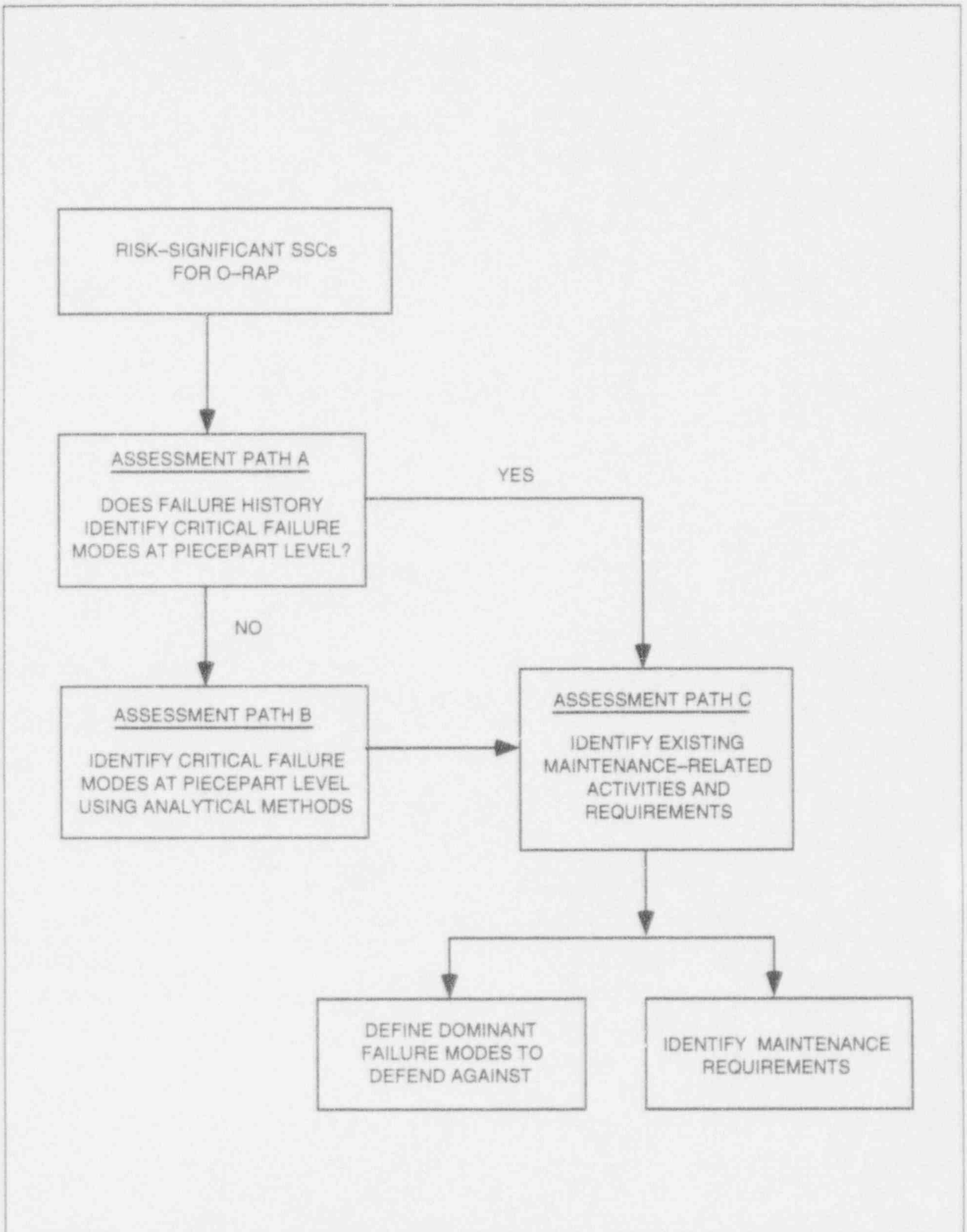


Figure 17.3-2
Process for Determining Dominant Failure Modes of Risk-Significant SSCs

~~**Figure 17.3-3**~~
~~**Process for Determining Dominant Failure Modes of Risk-Significant SSCs**~~

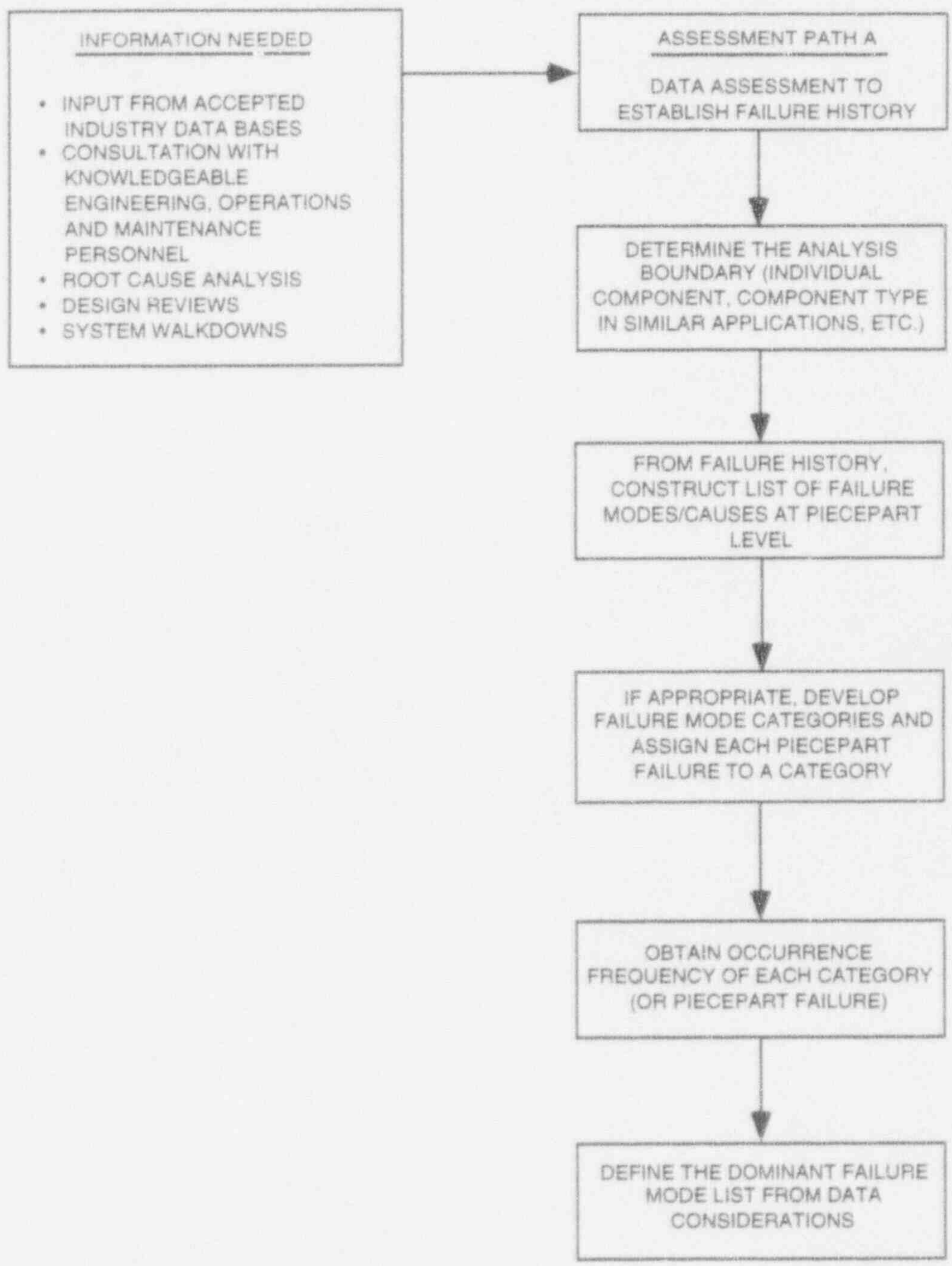


Figure 17.3-3 Use of Failure History to Define Failure Modes

~~**Figure 17.3-4 Use of Failure History to Define Failure Modes**~~

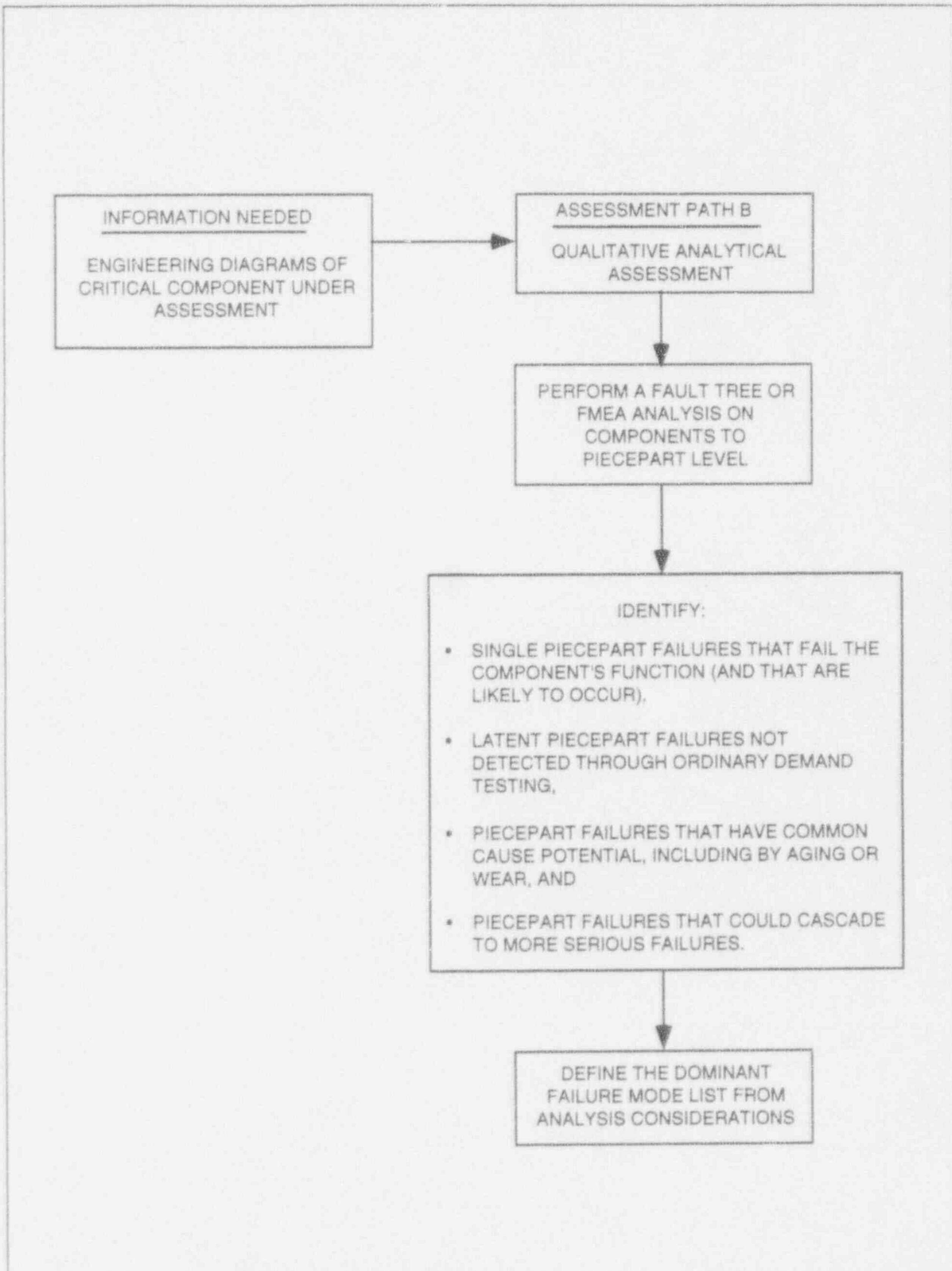


Figure 17.3-4 Analytical Assessment to Define Failure Modes

~~**Figure 17.3-5 Analytical Assessment to Define Failure Modes**~~

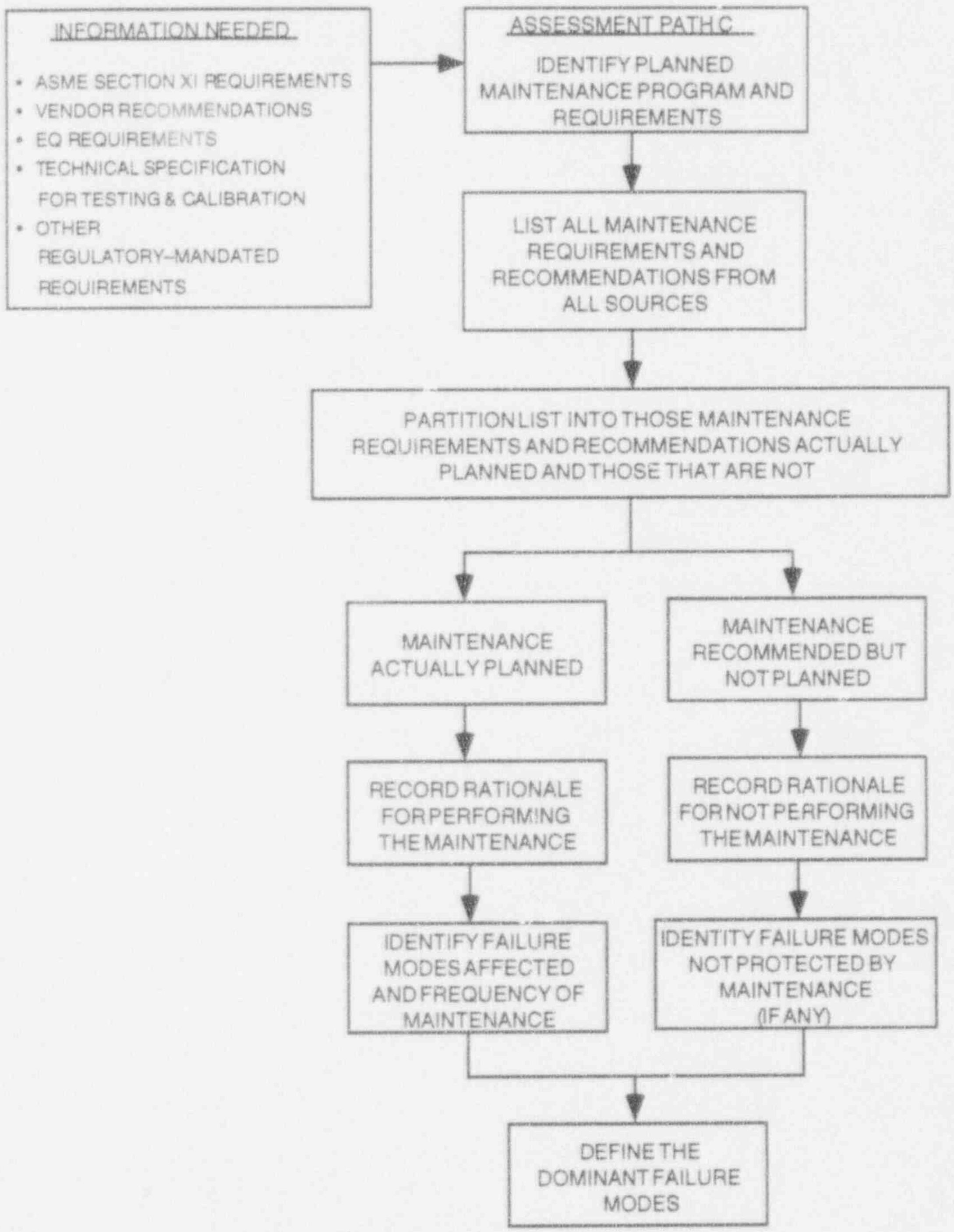


Figure 17.3-5
Inclusion of Maintenance Requirements in the Definition of Failure Modes

~~**Figure 17.3-6**~~
~~**Inclusion of Maintenance Requirements in the Definition of Failure Modes**~~

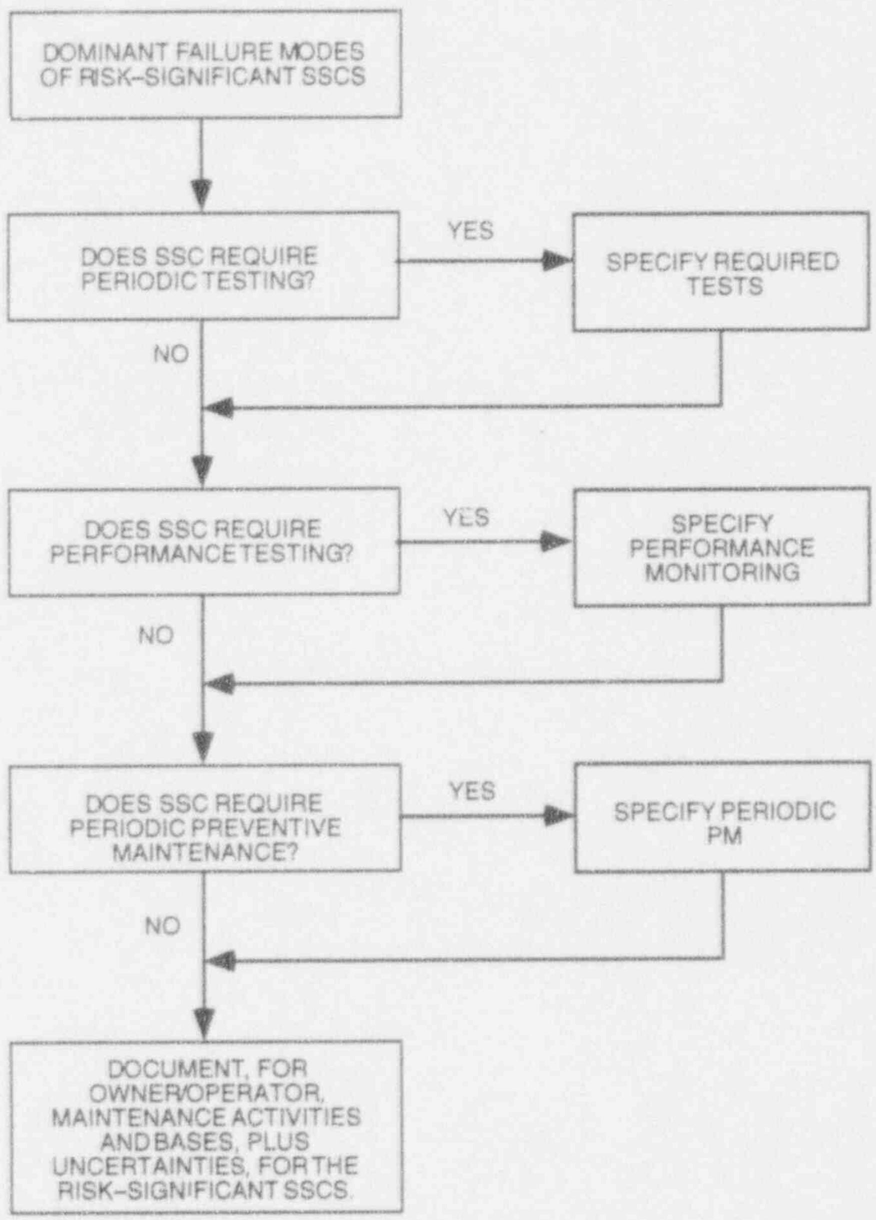


Figure 17.3-6 Identification of Risk-Significant SSC O-RAP Activities (Example)

~~Figure 17.3-7 Identification of Risk-Significant SSC O-RAP Activities (Example)~~

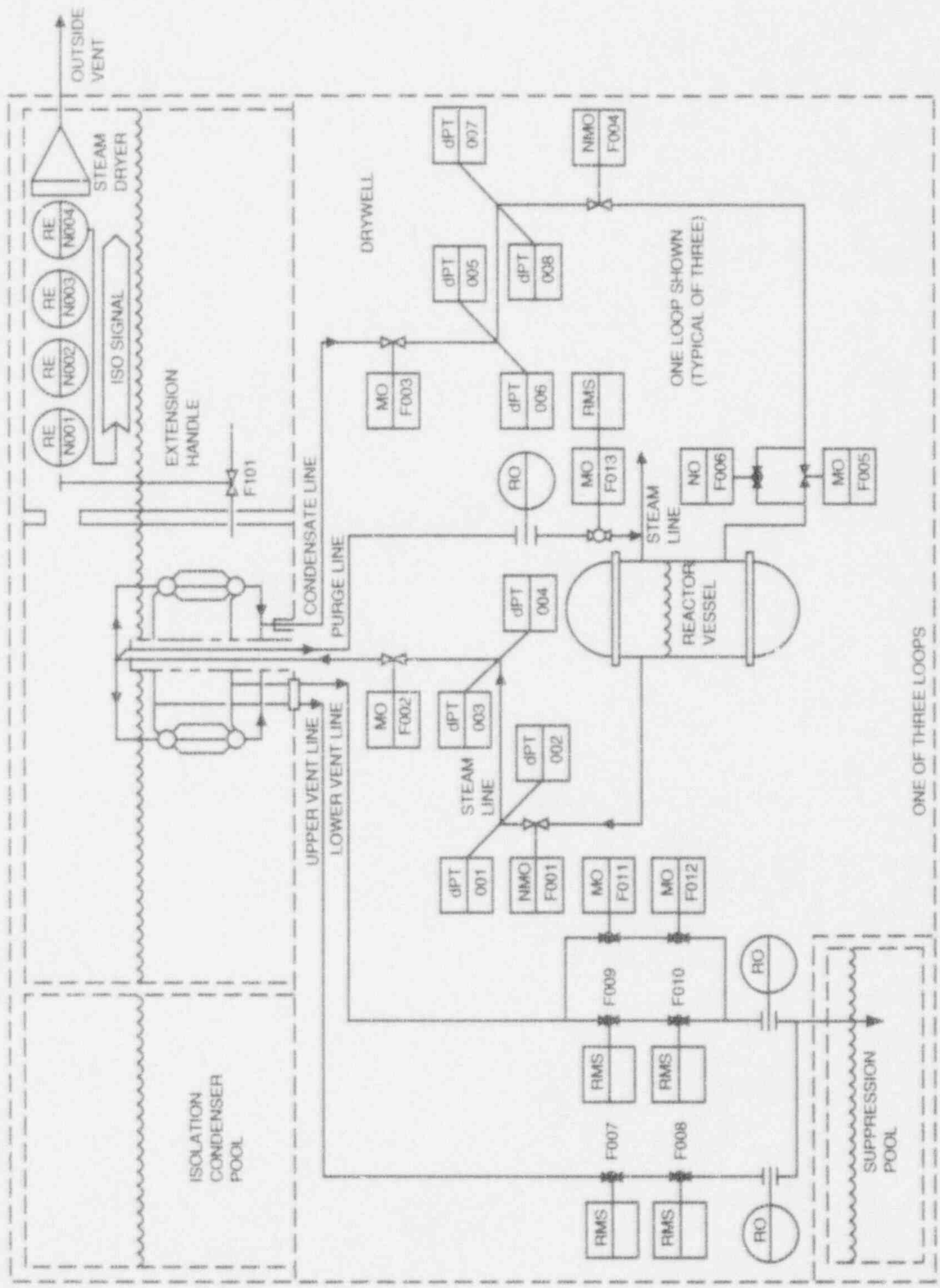
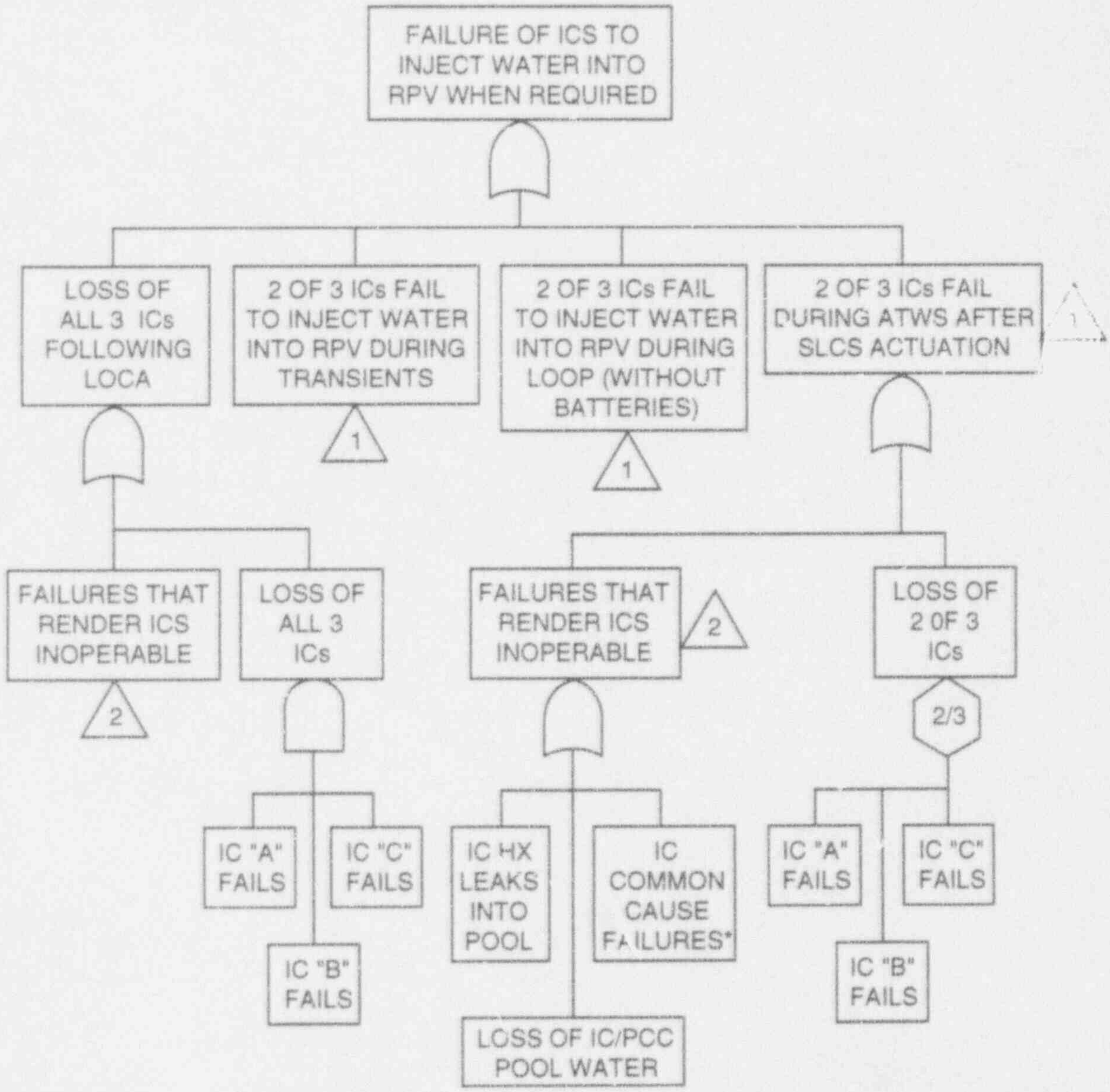


Figure 17.3-7 Isolation Condenser System P&ID (Simplified)

~~Figure 17.3-8 Isolation Condenser System P&ID (Simplified)~~



* SOME OF THE COMMON CAUSE FAILURES ARE DIFFERENT FOR DIFFERENT ACCIDENT SCENARIOS.

<u>Figure 17.3-8 Example Isolation Condenser System Top Level Fault Tree</u>

~~Figure 17.3-9 Example Isolation Condenser System Top Level Fault Tree~~

RAI Number: RPEB.10

Question:

SSAR Section 17.3.4 states a major factor in plant reliability assurance is risk-focused maintenance. However, the description appears to be limited to safety-related maintenance and not risk-focused maintenance. GE should clarify what is meant by risk-focused maintenance in SSAR Section 17.3.4.

GE Response:

"Risk-focused maintenance" is defined in Section 17.3.4. This definition is from NUREG/CR-5695, "A Process for Risk Focused Maintenance." Also refer to Section 17.3.1.

RAI Number: RPEB.11

Question:

SSAR Section 17.3.5 refers to Figure 17.3-1, "Typical GE-NE Organizational Chart for an SBWR Project." The staff noted this organizational chart differs from the chart provided in the ABWR SSAR and pertains only to the GE-NE portion of the D-RAP. The section also describes the correct D-RAP organization in the future tense. GE should: (1) state that a combined operating license applicant will need to supply a D-RAP organization description at the time of application for those risk-significant SSCs that are designed or procured by the applicant; (2) clarify the differences between the ABWR and SBWR D-RAP organizations; and (3) use the present tense to describe the GE-NE D-RAP organization that is currently in place.

GE Response:

The organization description of Section 17.3.5 and Figure 17.3-1 will be rewritten to be consistent with the ABWR SSAR and to address the comments above. (See Amendment 1 to the SSAR attached to RAI RPEB.9.)

RAI Number: RPEB.12

Question:

SSAR Section 17.3.7 states the reliability of risk-significant SSCs, which are identified by the PRA, will be evaluated at the detailed design stage by appropriate design reviews and reliability analyses. GE should clarify the meaning of "detailed design stage" and indicated if it is before or after FDA. While the use of PRA to determine risk-significant SSCs is preferred, there are systems or events (e.g., fires) where use of importance measures are limited by the level of detail in the PRA models. Therefore, GE should expand its definition of ways of identifying risk-significant SCCs to include the use of deterministic or other methods.

GE Response:

Risk-significant SSCs are "indicated by PRA or other sources," in Section 17.3.7. Please refer to Section 17.3.6 regarding "sources other than the PRA."

RAI Number: RPEB.14

Question:

SSAR Section 17.3.10 outlines portions of a referencing applicant's O-RAP. The O-RAP will have various programmatic interfaces that are listed in this section including procurement of replacement equipment. However, the initial equipment procurement done by the combined operating license applicant is not addressed. GE should include both initial and replacement equipment procurement in the list of programmatic interfaces in SSAR Section 17.3.10.

GE Response:

Section 17.3.10 will be revised to include both initial equipment procurement and replacement equipment procurement. (See Amendment 1 to the SSAR attached to RAI RPEB.9.)

RAI Number: RPEB.15

Question:

SSAR Section 17.3.11.4 describes the identification of risk-significant SSCs. However, Table 17.3-1, "ICS Components With Largest Contribution to Core Damage Frequency," is not referred in this section or any other section in Chapter 17.3. Also, SSAR Table 17.3-1 must include risk importance measures (Risk Achievement and Fussell-Veseley) associated with the components listed in the table. GE should reference Table 17.3-1 in the text of the SSAR and should reference or discuss the associated importance measures of the component's contribution to core damage frequency.

GE Response:

Table 17.3-1 will be revised to include Fussell-Veseley values for components listed. Table 17.3-1 will be referenced in Sections 17.3.11.4 and 17.3.11.5. (See Amendment 1 to the SSAR attached to RAI RPEB.9.)

RAI Number: RPEB.16

Question:

SSAR Section 17.3.11.5 refers to components in Table 17.3-2 as having high importance and uses that result to show how Figure 17.3-2 does not provide a relative measures of the components contribution to core damage frequency. As stated above, Table 17.3-1 also lacks such a measure. GE should provide some relative risk importance measure in these tables so that the system design response argument can be more easily followed.

GE Response:

In Amendment 1 of the SSAR, Table 17.3-1 will have Fussell-Vaseley importance values added to the Risk Achievement worth of the valves listed in Table 17.3-1 and included in Table 17.3-2. (See Amendment 1 attached to RAI RPEB.9.)

RAI Number: SPLB.1

Question:

In Section 3.11.1 of the SBWR SSAR it is stated that "a list of all 10 CFR 50.49(b) electrical and safety-related mechanical equipment that is located in a harsh environment area will be included in the Environmental Qualification Document (EQD) to be prepared as mentioned in Subsection 3.11.4."

The staff finds it acceptable to qualify both electrical and mechanical equipment in accordance with the requirements of 10 CFR 50.49. However, 10 CFR 50.49 is not a requirement for environmental qualification of mechanical equipment. There are no detailed requirements for environmental qualification of mechanical equipment; however, GDC 1, "Quality Standards and Records," GDC 4, "Environmental and Missile Design Bases," and Appendix B to 10 CFR Part 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Section III, "Design Control," and XVII, "Quality Assurance Records"), contain the following requirements related to equipment qualification:

- Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.

- Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety-related functions.

- Design control measures shall be established for verifying the adequacy of design.

- Equipment qualification records shall be maintained and shall include the results of tests and material analyses.

For mechanical equipment, the staff review will concentrate on materials (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, diaphragms, etc.) which are sensitive to environmental effects. Your review and evaluation should include the following:

1) Identification of safety-related mechanical equipment located in harsh environmental areas, including required operating time.

2) Identification of non-metallic subcomponents of this equipment.

3) Identification of the environmental conditions this equipment must be qualified for. The environments defined in the electrical equipment program are also applicable to mechanical equipment.

4) Identification of non-metallic material capabilities.

5) Evaluation of environmental effects.

If it is decided that environmental qualification of mechanical equipment will be in accordance with 10 CFR 50.49, as is currently indicated in SBWR SSAR Section 3.11.1, then the electrical equipment and the mechanical equipment must be identified as a separate groups.

GE Response:

The qualification program to be applied to the SBWR mechanical equipment will use applicable portions of the NRC approved Licensing Topical Report (LTR) NEDE-24326-1-P, which complies (as noted in the reference below) with 10 CFR 50.49, Appendix A to 10 CFR 50 (GDC 1, 2, 4 and 23), and Appendix B to 10 CFR 50 (Sections III and XI). Thus, the program for safety-related mechanical equipment in a harsh environment complies with all five requirements addressed specifically in the question. The electrical and the mechanical equipment are qualified as separate groups; metallic pressure boundary of mechanical equipment is considered qualified by the application of an ASME stamp and nonmetallic materials (e.g., seals, gaskets, lubricants, hydraulic system fluids, diaphragms, etc.) are shown to be capable of maintaining their capabilities during their life.

Reference: NRC Memorandum for Frank J. Miraglia from James P. Knight, "SAFETY EVALUATION REPORT (SER) FOR GENERAL ELECTRIC QUALIFICATION PROGRAM, NEDE-24326-1-P," dated July 1, 1983.

RAI Number: SPLB.2

Question:

In Section 3.11.2.1 the radiation source term used in th accident analysis must be identified (e.g., will TID-14844 be used in accordance with guidan e of NUREG-0588 and RGs 1.3 and 1.4).

GE Response:

The SSAR Subsections 3.11.2.1 and 3.11.2.3 and Appendix 3D (footnotes of Tables 3D-6 through 3D-9 and of Tables 3D-14 through 3D-17) will be revised in Amendment 1(see attached) to specify that the radiation sources associated with the design basis accident (DBA) and based on NUREG-1465 will be used for the DBA radiation (gamma and beta) environmental conditions for equipment qualification (see attached). The basis for sources will not use TID-14844 (or its associated regulatory guides or SRPs) or refer to SSAR Chapter 12. A basis that is acceptable to the NRC and consistent with the June 1992 draft of NUREG-1465 will be used for SBWR equipment qualification.

Safety-related mechanical equipment and 10CFR49(b) electrical equipment located in a harsh environment must perform its proper safety function in environments during normal, abnormal, test, design basis accident and post-accident conditions as applicable. A list of all 10CFR49(b) electrical and safety-related mechanical equipment that is located in a harsh environment area will be included in the Environmental Qualification Document (EQD) to be prepared as mentioned in Subsection 3.11.4.

3.11.2 Environmental Conditions

3.11.2.1 General Requirements

Environmental conditions for the zones where safety-related equipment is located are calculated for normal, abnormal, test, accident and post-accident conditions and are documented in Appendix 3D, Equipment Qualification Environmental Design Criteria (EQEDC). Environmental conditions are tabulated by zones contained in the referenced building arrangements. Typical equipment in the noted zones is shown in the referenced system P&ID and IED design drawings.

~~Environmental parameters include temperature, pressure, relative humidity, and neutron dose rate and integrated dose. Radiation dose for gamma and beta data for both normal and accident conditions will be provided by the COL applicant referencing the SBWR design in accordance with the requirements in Subsection 12.2.3.1. The radiation requirements are hardware specific owing to the need to model the specific equipment to be deployed. Where applicable, these parameters are given in terms of time based profiles.~~

Environmental parameters include thermodynamic parameters (temperature, pressure and relative humidity), radiation parameters (dose rates and integrated doses of neutron, gamma and beta exposure) and chemical spray parameters (chemical composition and the resulting pH). Subsection 3.11.2.3 describes further the chemical and radiation environments.

The magnitude and 60-year frequency of occurrence of significant deviations from normal plant environments in the zones have insignificant effects on equipment total thermal normal aging or accident aging. Abnormal and test condition environments are overshadowed by the normal or accident conditions according to the Appendix 3D tables.

Margin is defined as the difference between the most severe specified service conditions of the plant and the conditions used for qualification. Margins shall be included in the qualification parameters to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance. The environmental conditions shown in the Appendix 3D tables do not include margins.

Radiation Environment

Safety-related systems and components are designed to perform their safety-related function when exposed to the normal operational radiation levels and accident radiation levels.

The normal operational exposure is based on the radiation sources provided in Chapter 12.

~~Radiation sources associated with the DBA and developed in accordance with NUREG-0588 (Reference 3.11-2) are provided in Chapter 15.~~

~~Integrated doses associated with normal plant operation and the design basis accident condition for various plant compartments are described in Appendix 3D.~~

The radiation sources associated with the design basis accident (DBA) and developed in accordance with NUREG-1465 are used. Dose rates and integrated doses of neutron, gamma and beta radiation that are associated with normal plant operation and the DBA condition for various plant compartments are presented in Appendix 3D; these parameters are presented in terms of time-based profiles where applicable.

The gamma and beta doses in Appendix 3D are bounding values based on generic design considerations, and are to be revised and/or verified by the COL applicant based upon the site-specific equipment considerations (exact design, specific location, materials of construction and leakage characteristics).

3.11.3 Qualification Program, Methods and Documentation

10CFR49(b) electrical equipment that is located in a harsh environment is qualified by test or other methods as described in IEEE 323 and permitted by 10CFR50.49(f) (Reference 3.11-1). Equipment type test is the preferred method of qualification.

Safety-related mechanical equipment that is located in a harsh environment is qualified by analysis of materials data which are generally based on test and operating experience.

The qualification program and methodology are described in detail in the NRC approved licensing Topical Report on GE's environmental qualification program (Reference 3.11-3). This report also addresses compliance with the applicable portions of the General Design Criteria of 10CFR50, Appendix A, and the Quality Assurance Criteria of 10CFR50, Appendix B. Additionally, the report describes conformance to NUREG-0588 (Reference 3.11-2), and Regulatory Guides and IEEE Standards referenced in SRP 3.11.

Mild environment is that which, during or after a design basis event (DBE, as defined in Reference 3.11-3), would at no time be significantly more severe than that existing during the normal, test and abnormal events.

Table 3D-6 Radiation Environment Conditions Inside Containment Vessel for Normal Operating Conditions

Plant Zone/Typical Equipment	Operating Dose Rate* †		Integrated Dose† ‡	
	Gamma (R/h)	Beta(R/h)	Gamma(R)	Beta(R)
(b-1) Upper drywell [Figs 1.1-1 and 21.1.2-2 Sh1&2]				
(b-2) Upper area of lower drywell [Figs 1.1-1 and 21.1.2-2 Sh1&2]				
(b-3) Lower area of lower drywell [Figs 1.1-1 and 21.1.2-2 Sh1&2]				
(b-4) Suppression Chamber - Suppression pool and gas space [Figs 6.2-5 and 21.1.2-2 Sh1&2]				

* Operating dose rate is at 100% rated power and away from radiation source.

† The doses ~~will be provided when acceptable radiation source terms are defined~~are based on the radiation sources provided in Chapter 12.

‡ Integrated dose means the integrated value over 60 years.

* Operating dose rate is at 100% rated power and away from radiation source.

† The doses ~~will be provided when acceptable radiation source terms are defined~~are based on the radiation sources provided in Chapter 12.

‡ Integrated dose means the integrated value over 60 years.

Table 3D-8 Radiation Environment Conditions Inside Reactor Building for Normal Operating Conditions

Plant Zone/Typical Equipment	Operating Dose Rate* †		Integrated Dose† ‡	
	Gamma (R/h)	Beta(R/h)	Gamma(R)	Beta(R)
MS and FW Tunnel MSL isolation valve MSL drain isolation valve Feedwater isolation valve [Figs 5.4-2, 21.1.2-2 Sh 7]				
ICS condenser and piping outside containment [Fig 21.5.4-1]				
FAPCS emergency makeup water lines [Figs 21.9.1-1, 21.9.1-2, and 21.1.2-2 Sh 13]				

* Operating dose rate is at 100% rated power and away from radiation source.

† The doses ~~will be provided when acceptable radiation source terms are defined~~are based on the radiation sources provided in Chapter 12.

‡ Integrated dose means the integrated value over 60 years.

Table 3D-9 Radiation Environment Conditions Inside Control Envelope for Normal Operating Conditions

Plant Zone/Typical Equipment	Operating Dose Rate* †		Integrated Dose† ‡	
	Gamma (R/h)	Beta(R/h)	Gamma(R)	Beta(R)
Main control room panels [Fig 21.1.2-2 Sh 13]				
Emergency breathing air system [Fig 21.6.4-1]				

* Operating dose rate is at 100% rated power and away from radiation source.

† The doses ~~will be provided when acceptable radiation source terms are defined~~are based on the radiation sources provided in Chapter 12.

‡ Integrated dose means the integrated value over 60 years.

Table 3D-14 Radiation Environment Conditions Inside Containment Vessel for Accident Conditions

Plant Zone/Typical Equipment	Operating Dose Rate* †		Integrated Dose† ‡	
	Gamma (R/h)	Beta (R/h)	Gamma (R)	Beta (R)
(b-1) Upper drywell [Figs 1.1-1 and 21.1.2-2 Sh1&2]				
(b-2) Upper area of lower drywell [Figs 1.1-1 and 21.1.2-2 Sh1&2]				
(b-3) Lower area of lower drywell [Figs 1.1-1 and 21.1.2-2 Sh1&2]				
(b-4) Suppression Chamber - Suppression pool and gas space [Figs 6.2-5 and 21.1.2-2 Sh1&2]				

* ~~Assumes that 100% of the inert gases, 50% of Halogen, and 1% of the solid fission products are released from the core during LOCA~~The radiation sources developed in accordance with NUREG-1465 are used.

† ~~The gamma and beta doses will be provided by the applicant referencing the SBWR design in accordance with the requirements of Section 12.2.3~~The gamma and beta doses are bounding values based upon generic design considerations, and are to be revised and/or verified by the COL applicant based upon the site-specific equipment considerations (exact design, specific location, materials of construction and leakage characteristics.

‡ Integrated dose is for 6 months.

* ~~Assumes that 100% of the inert gases, 50% of Halogen, and 1% of the solid fission products are released from the core during LOCA~~The radiation sources developed in accordance with NUREG-1465 are used.

† ~~The gamma and beta doses will be provided by the applicant referencing the SBWR design in accordance with the requirements of Section 12.2.3~~The gamma and beta doses are bounding values based upon generic design considerations, and are to be revised and/or verified by the COL applicant based upon the site-specific equipment considerations (exact design, specific location, materials of construction and leakage characteristics.

‡ Integrated dose is for 6 months.

Table 3D-16 Radiation Environment Conditions Inside Reactor Building for Accident Conditions

Plant Zone/Typical Equipment	Operating Dose Rate* †		Integrated Dose† ‡	
	Gamma (R/h)	Beta (R/h)	Gamma (R)	Beta (R)
MS and FW Tunnel MSL isolation valve MSL drain isolation valve Feedwater isolation valve [Figs 5.4-2, 21.1.2-2 Sh 7]				
ICS condenser and piping outside containment [Fig 21.5.4-1]				
FAPCS emergency makeup water lines [Figs 21.9.1-1, 21.9.1-2, and 21.1.2-2 Sh 13]				

* ~~Assumes that 100% of the inert gases, 50% of Halogen, and 1% of the solid fission products are released from the core during LOCA~~The radiation sources developed in accordance with NUREG-1465 are used.

† ~~The gamma and beta doses will be provided by the applicant referencing the SBWR design in accordance with the requirements of Section 12.2.3~~The gamma and beta doses are bounding values based upon generic design considerations, and are to be revised and/or verified by the COL applicant based upon the site-specific equipment considerations (exact design, specific location, materials of construction and leakage characteristics.

‡ Integrated dose is for 6 months.

Table 3D-17 Radiation Environment Conditions Inside Control Room Envelope for Accident Conditions

Plant Zone/Typical Equipment	LOCA* †		Integrated Dose† ‡	
	Gamma (R/h)	Beta (R/h)	Gamma (R)	Beta (R)
Sealed Emergency Operating Area Main control room panels [Fig 21.1.2-2 Sh 13]				
Emergency breathing air system (EBAS) [Fig 21.6.4-1]				

* ~~Assumes that 100% of the inert gases, 50% of Halogen, and 1% of the solid fission products are released from the core during LOCA~~The radiation sources developed in accordance with NUREG-1465 are used.

† ~~The gamma and beta doses will be provided by the applicant referencing the SBWR design in accordance with the requirements of Section 12.2.3~~The gamma and beta doses are bounding values based upon generic design considerations, and are to be revised and/or verified by the COL applicant based upon the site-specific equipment considerations (exact design, specific location, materials of construction and leakage characteristics.

‡ Integrated dose is for 6 months.

RAI Number: SPLB.3

Question:

Confirmed that the environmental qualification records discussed in SBWR SSAR Section 3.11.4 will be in accordance with requirements of 10 CFR 50.49(j).

The following 21 RAIs refer to Section 3.4.1, Flood Protection.

GE Response:

The paragraph, Environmental Qualification Records, in SSAR Subsection 3.11.4 will be revised in Amendment 1 of the SSAR (see attached) as follows: "The results of the qualification tests shall be recorded and maintained in an auditable file in accordance with requirements of 10 CFR 50.49(j)."

The vendors of equipment located in a mild environment are required to submit a certificate of compliance certifying that the equipment has been qualified to assure its required safety-related function in its applicable environment. This equipment is qualified for dynamic loads as addressed in Sections 3.9 and 3.10. Further, a surveillance and maintenance program will be developed to ensure the operability during its designed life.

The procedures and results of qualification by tests, analyses or other methods for the safety-related equipment will be documented, maintained, and reported as mentioned in Subsection 3.11.4. The requirements for this documentation are presented in GE's environmental qualification program (Reference 3.11-3).

3.11.4 COL License Information

Environmental Qualification Document

The EQD shall be prepared summarizing the qualification results for all equipment identified in Subsection 3.11.1. The EQD shall include the following:

- The test environmental parameters and the methodology used to qualify the equipment located in harsh environments shall be identified.
- A summary of environmental conditions and qualified conditions for the equipment located in a harsh environment zone shall be presented in the system component evaluation work (SCEW) sheets as described in Table I-1 of GE's environmental qualification program (Reference 3.11-3). The SCEW sheets shall be compiled in the EQD.

Environmental Qualification Records

The results of the qualification tests shall be recorded and maintained in an auditable file in accordance with requirements of 10 CFR 50.49 (j).

3.11.5 References

3.11-1 Code of Federal Regulations, Title 10, Chapter I, Part 50, Paragraph 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant.

3.11-2 Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, NUREG-0588.

3.11-3 General Electric Environmental Qualification Program, NEDE-24326-1-P, Proprietary Document, January 1983.

RAI Number: SPLB.4

Question:

Identify all safety-related equipment and equipment important-to-safety (ITS) (i.e., non-safety related equipment whose failure could adversely affect the ability of safety-related equipment to perform its safety function) requiring protection from internal and external flooding.

GE Response:

There is no non-safety-related equipment whose failure, due to either external flooding or internal flooding resulting from a moderate energy pipe failure, could adversely affect the ability of safety-related equipment to perform its safety function.

All safety-related equipment is located within the containment, safety envelope, steam tunnel, and isolation condenser/passive containment cooling (IC/PCC) pool. These areas are protected from external flooding by the Reactor Building, which is sealed to Elevation 10000 mm, which is 0.3048 meter (1 foot) above the site flood elevation. Any potential for flooding due to failure of external tanks or basins will be prevented by ensuring the resulting flood water will be directed away from the Reactor Building by site drainage characteristics. Flooding of the Reactor Building from other buildings via any connecting tunnels will be prevented by provision of watertight barriers where required.

Safety-related equipment within the containment is qualified for loss-of-coolant-accident (LOCA) conditions which will envelope the conditions for moderate energy pipe failure. Safety-related equipment within the steam tunnel is qualified for main steam line breaks and feedwater line breaks in the steam tunnel. Water will drain into the turbine building and away from other safety-related areas.

The floor drainage system for the Reactor Building, both within and outside the safety envelope, for spaces above Elevation -6400 mm, is sized to discharge the water from the maximum postulated moderate energy line break or from fire fighting activities, thereby limiting the height of water buildup. Equipment will be installed above the maximum calculated flood height.

All moderate energy pipe break leakage or fire fighting water discharged in the Reactor Building will be drained to the sumps on Elevation -6400 mm. These sumps and the drainage to them are hydraulically separate so that flooding from spaces containing one equipment train or division will not flood a redundant train or division via the drainage system.

In the analysis, the sump pumps are conservatively assumed to be out of service. All leakage drained to or originating on Elevation -6400 mm is distributed over the area connected to the associated sump by the floor drainage system. Drainage is arranged so that all drainage from "A" train of non-safety-related equipment rooms and "A" and "C" divisions of safety-related equipment rooms drains to one sump and drainage from "B" train of non-safety-related equipment rooms and "A" and "C" division safety-related equipment rooms drains to the other sump. Each sump also is connected to two of the four safety-related division rooms within the safety envelope on Elevation -6400 mm. An isolation valve will isolate each safety-related division room on rising water level to prevent back flooding through the drainage system from outside the safety envelope.

RAI Number: SPLB.5

Question:

Provide a flood analysis that identifies potential sources of internal flooding on a floor-by-floor basis in all buildings containing safety-related equipment. How will safety-related equipment and equipment ITS be protected from flooding from these sources?

GE Response:

Section 19CC.4 of Appendix 19C, submitted in February 1993, provides a description of the deterministic flood analysis performed to demonstrate that moderate energy pipe failures will not prevent safe shutdown and provides a description of the plant features protecting safety-related equipment from internal sources of flooding.

RAI Number: SPLB.6

Question:

Identify which safe shutdown equipment will be located above the maximum flood height and which will be qualified for flooded conditions.

GE Response:

All equipment required for safe shutdown, except for the Isolation Condenser System condensers and Passive Containment Cooling System condensers, will be located above the maximum water accumulation height resulting from failure of a moderate energy pipe or fire fighting activities, as indicated in the SPLB.4 and SPLB.5 responses. The maximum height due to moderate energy pipe failures in the Reactor Building is less than 0.3048 meter (1 foot) and no accumulations due to external sources are credible for the reasons stated in the response to SPLB.4.

RAI Number: SPLB.7

Question:

Discuss the ability of safety-related equipment to perform its safety function while fully flooded, partially flooded, or wet (e.g., from spray)?

GE Response:

Safety-related equipment is designed to operate when flooded; protected from partial flooding, complete flooding, or wetting; or not required to function when partially flooded, completely flooded, or wetted. All safety-related components located inside the containment are designed to perform their safety-related functions in the required environment, including fully flooded, partially flooded, or wet condition, as applicable. Safety-related equipment outside the containment will be located above the highest flood level resulting from a moderate energy pipe break or fire fighting activities. Also, it may be possible for safety-related equipment outside the containment to be wetted by spray from failed moderate energy piping or fire fighting activities. However, only one safety-related division is expected to be affected during any postulated event. Once all piping and equipment is located, a review will be performed to identify safety-related equipment required to remain functional and vulnerable to spray from postulated moderate energy pipe failures. Such vulnerabilities will be addressed by showing the safety-related equipment can withstand the spray condition, moving the safety-related equipment, moving the pipe, and/or adding spray shields to eliminate the spray concern. This review will be done by GE as part of the standard plant design.

RAI Number: SPLB.8

Question:

The SSAR states that exposure to water spray will be evaluated once equipment locations and piping routings are finalized. Who will perform this evaluation, GE, or the COL applicant?

GE Response:

As indicated in the response to SPLB.7 it is intended that GE will evaluate the exposure of safety-related equipment to spray from failed moderate energy piping which is part of the standard design and from fire fighting activities to ensure the acceptability of the standard design.

RAI Number: SPLB.9

Question:

Is flooding associated with the break of a high-energy line considered in the flood analysis?

GE Response:

Section 3.4 did not consider flooding due to high-energy pipe failure. However, there will be no adverse flooding of safety-related equipment in the Reactor Building due to high-energy line breaks. Three high-energy line break scenarios have been evaluated. Flooding due to a main steam or feedwater line break in the steam tunnel is contained in the steam tunnel and directed into the Turbine Building away from safety-related equipment. Normally closed isolation valves in the steam tunnel drains prevent water from flooding from the steam tunnel to the Reactor Building through the drains. Flooding due to a reactor water cleanup/shutdown cooling (RWCU/SDC) line break outside of containment is confined to areas of the Reactor Building which do not contain safety-related equipment and is enveloped by moderate energy line breaks. Flooding due to an isolation condenser (IC) line break outside of containment will be contained in the isolation condenser/passive containment cooling (IC/PCC) pool and will not flood the Reactor Building.

RAI Number: SPLB.10

Question:

Is separation of equipment utilized as a means of flood protection? If so, which safety-related systems utilize separation to provide flood protection?

GE Response:

Separation of equipment to protect against the effect of flooding is not specifically done, although equipment may be separated for other reasons. Leakage from a moderate energy pipe failure or water from fire fighting activities in the Reactor Building is removed from floor slabs at all elevations above -6400 mm by the floor drain system. The fire protection separation barriers between the safety-related divisions within the safety envelope provide some protection against internal flooding affecting more than one division. However, the walls are only required to withstand a maximum of 0.3048 meter (1 foot) water. The fire barriers forming the floor and walls of the steam tunnel are also designed as watertight barriers. By not separating the various areas for flood control, the largest number of floor drains will be available for leakage removal to effectively limit the accumulation level. The various internal equipment compartment floors and walls, however, will be effective in limiting extent of spray.

On Elevation -6400 mm of the Reactor Building, the drains from the four safety division equipment rooms are separated so that two flow to one sump and the remaining two flow to the other sump. The leakage accumulation, when collected in one of the sumps, can only flow back through the connecting floor drain system to the associated non-safety-related train equipment rooms on Elevation -6400 mm and to two of the four safety division equipment division rooms, if the sump pumps are out of service. Under these conditions, the maximum level of accumulation will be less than 0.3048 meter (1 foot) and all electrical equipment will be mounted above that level. As further assurance that safety-related equipment in the safety envelope will not be damaged by back flooding, an isolation valve, which will close on a rising water level, is provided in the drain line from each safety division at Elevation -6400 mm to avoid back flooding the safety division through the interconnected floor drain system.

RAI Number: SPLB.12

Question:

Are any internal passageways too large to close with a single door? If so, how will leakage be prevented?

GE Response:

No internal passageways too large for closure with a single door or requiring leak-tight closure are currently required to accommodate flooding in the Reactor Building from either a moderate energy pipe failure or fire fighting activities.

RAI Number: SPLB.13

Question:

Provide design information on water seals, waterstops, watertight doors, and other protective features.

GE Response:

Detailed design of waterseals, waterstops, and watertight doors will be of standardized design for such features. Reactor Building moderate energy pipe failure leakage will flow via the floor drains serving the leak area to Elevation -6400 mm. Each sump and the associated drains are hydraulically separated from the other sump and drains. The compartments served by these drainage systems are separated from each other and common corridors by suitably rated, commercially available, watertight doors. Piping, wiring, ducting, etc., that penetrate the walls of these compartments will generally be above the anticipated flood water level. If they are located below the anticipated flood level, they will be adequately sealed.

RAI Number: SPLB.14

Question:

Identify all monitors which detect flooding in areas containing safety-related equipment and equipment ITS safety related. Are these monitors safety related?

GE Response:

All leakage flows to the Reactor Building sumps at Elevation -6400 mm. The level instrumentation in these sumps, which are not safety related, will provide appropriate leak detection. These monitors have sufficient redundancy and are normally in service during plant operation to provide a reliable indication of water accumulation in the sumps. The high-high alarms are not typically expected to activate during normal operation and will provide an indication that the sump pumps cannot control the drainage flow. Additionally, signals from the Radwaste Collection System will indicate excessive sump discharge volume.

A water-level sensor in each safety-related division in the safety envelope at Elevation -6400 mm will detect rising water level and isolate the associated drain for that division to prevent back flooding through the drain lines from affecting more than one division. These level detectors are safety related.

RAI Number: SPLB.15

Question:

Do any open-cycle systems enter any buildings housing safety- related equipment and equipment ITS? If so, how will this equipment be protected from the effects of a break in that part of the open-cycle system within the building?

GE Response:

In the standard design, there are no open-cycle systems. However, since portions of the Plant Service Water System (PSWS) are site specific, the combined operating license (COL) applicant could elect to have it become an open-cycle system. In either case, flooding is controlled in the same manner.

The PSWS enters the Reactor Building. The safety-related equipment is protected since it is installed above maximum calculated flood height and is isolated from the postulated flood plain. Additionally, shutoff valves are provided exterior to the building so the pipe inventory of water can be isolated in the event of a crack in the pipe within the Reactor Building.

RAI Number: SPLB.16

Question:

How will safety-related equipment and equipment ITS be protected from failures of structures, systems, and components which are not within the SBWR design scope?

GE Response:

There are no failures of structures, systems, and components outside the SBWR standard design scope from which safety-related equipment and equipment important to safety (ITS) require protection from flooding provided that the site meets the design parameters and the site contours are controlled by the combined operating license (COL) applicant to ensure that potential flooding is directed away from the Reactor Building.

RAI Number: SPLB.17

Question:

How will the remote shutdown panel (RSP) be protected from external and internal flooding?

GE Response:

The remote shutdown panel is located at Elevation +10000 mm in the Reactor Building. The Reactor Building is not subject to external flooding, above Elevation +9695.2 mm. Also, the maximum flood level in the Reactor Building, due to moderate energy pipe failure or fire fighting activities, is less that 0.3048 meter (1 foot). This panel will be mounted at least 0.3048 meter (1 foot) above Elevation +10000 mm.

RAI Number: SPLB.18

Question:

The SSAR states that the drain collection system and sumps are designed and separated so that drainage from a flooding compartment containing equipment for a train or division does not flow to compartments containing equipment for another system train or division. Provide design details of the drain collection system and sumps.

GE Response:

The sumps are hydraulically separated by partitions so that the highest water level that is calculated to occur at Elevation -6400 mm cannot overflow from one sump to another. The drain collection system is designed so that drainage from the "A" train equipment rooms outside the safety envelope and from the "A" and "C" safety-related division rooms inside the safety envelope are drained to one sump. The "B" train equipment rooms, the "B" and "D" safety-related division rooms, and the common corridors are drained to the second sump. Accordingly, if the sumps are out of service during a moderate energy pipe break or fire fighting event in the Reactor Building, all drainage will be directed to one of the two sumps. An isolation valve is provided to isolate each safety-related division drain on Elevation -6400 mm from its associated sump on rising water level so that back flooding into the safety-related division rooms from outside the safety envelope will not occur. It will be possible for all of the rooms at Elevation -6400 mm draining to one of the sumps to experience water accumulation due to backflow through the drainage system. However, since each of these rooms is fitted with watertight doors or isolated by valves and since the sumps are separated, accumulation of water in the train or divisions served by the other sump is not possible provided the water level stays below the calculated level.

RAI Number: SPLB.19

Question:

Identify whether flood protection depends upon the use of a dewatering system. If so, provide seismic, safety class, and quality group classifications.

GE Response:

Flood protection for the standard design SBWR does not depend on the use of a dewatering system. All safety-related equipment is in the Reactor Building. The Reactor Building grade elevation is above the site flood level, as are all exterior access openings. Exterior penetrations, walls, and floors below design flood and groundwater levels are sealed to withstand the hydrostatic pressure. Any seepage should be minimal and will be handled by the floor drainage collection system and discharged by the floor drain sumps. The floor drainage system and floor drain sumps are not seismic or safety grade systems. Any seepage will not have a significant effect on the volume of leakage during a Reactor Building moderate energy pipe failure or fire fighting event.

RAI Number: SPLB.20

Question:

Identify potential sources of external flooding from components which are within the SBWR design scope.

GE Response:

Reactor Building flooding from external sources is not considered viable since the plant grade is above site flood elevation. Also, the site grading and drainage, as well as sealing openings located below grade, will divert or provide barriers to site-generated flows from failed facilities. Typical of facilities which might have flooding potential if appropriate preventive design features were not provided are the circulating water cooling tower basin, Circulating Water System piping, plant service water cooling tower basin, plant service water piping, condensate storage tank, and makeup water storage tank.

RAI Number: SPLB.21

Question:

Identify safety-related equipment and equipment ITS which are subject to groundwater seepage, and discuss how this will be controlled.

GE Response:

All safety-related equipment located below grade is within the safety envelope portion of the Reactor Building. Groundwater seepage will be controlled by appropriate sealing of below-grade exterior penetrations, walls, and floors and by the Reactor Building floor drainage system. In the event that the Floor Drainage System cannot handle the seepage flow, the Safety Envelope Drainage System will be automatically isolated from the remainder of the Reactor Building Floor Drainage System on rising water level before the level in the safety envelope could threaten the operation of the safety-related equipment.

RAI Number: SPLB.22

Question:

Provide a discussion of possible flood hazards resulting from below-grade tunnels between buildings.

GE Response:

A tunnel connects the Reactor Building, which contains all of the safety-related equipment, and the Turbine Building. Flooding of the Reactor Building from the Turbine Building through the tunnel, due to failure at a circulating water expansion joint connection to the condenser water box or other Turbine Building flood sources, is prevented by providing suitable curbing at the tunnel entrance to divert the Turbine Building flooding away from the tunnel, and/or providing a watertight tunnel partition with sealed piping penetrations and watertight doors as required.

RAI Number: SPLB.23

Question:

SSAR Section 5.2.5 Reactor Coolant Pressure Boundary

Identify and describe the monitoring of all potential intersystem leakages that are not included in SSAR Section 5.2.5.2.2 subparagraph, "Intersystem Leakage Monitoring." Your response should include all the applicable (for the SBWR) systems and components connected to the reactor coolant system that are listed in Table 1 of SRP Section 5.2.5 and other systems that are unique to SBWR. Revise SSAR accordingly.

GE Response:

Section 5.2.5.2.2, subparagraph "Intersystem Leakage Monitoring" will be revised in Amendment 1 as follows:

"Intersystem leakage of radioactive material into each RCCWS train is monitored continuously by the PRMS. An in-line radiation monitor is provided at the RCCWS common discharge line that connects the cooling water output flows from the RWCU/SDC non-regenerative heat exchanger, the FAPCS heat exchanger, the upper and lower drywell coolers, the reactor building chiller, and the RCCWS air cooler. A high level of radioactivity is indicative of reactor coolant leakage into the closed loop RCCWS train. The high radiation level will be alarmed in the control room."

Isolation Condenser Radiation Leakage Monitoring

The vent discharge from each isolation condenser into the pool area is monitored separately for high radiation levels by the PRMS. Four divisional channels per isolation condenser are provided to sense for gamma radiation leakage using digital gamma sensitive detectors. A high radiation level will be annunciated in the main control room and will cause isolation of the defective isolation condenser.

Main Steamline Low Pressure Monitoring

The main steamline flow is monitored for low pressure by four pressure transmitters (two in each line) that sense the pressure downstream of the outboard MSIVs. The sensing points are located as close as possible to the turbine stop valves. A low steamline pressure can be an indication of a steamline leak or a malfunction of the reactor pressure control system. The isolation logic will automatically initiate closure of all MSIVs and the main steamline drain valves if pressure at the turbine falls below the setpoint during reactor operation.

Main Condenser Low Vacuum Monitoring

The pressure in the main condenser is monitored for low vacuum, which could indicate that primary reactor coolant is being lost through the main condenser. Four divisional pressure monitoring channels are provided to generate the trip on low vacuum level. The trip signal is used by the isolation logic for closure of the MSIVs and the steam drain line valves. The condenser vacuum measurement is bypassed during startup and shutdown operations to guard against unnecessary isolation.

Intersystem Leakage Monitoring

~~Intersystem radiation leakage into the RCCWS is monitored and analyzed by the PRMS. Liquid samplers are used to extract samples of the cooling water to such equipment as the non regenerative heat exchangers of the RWCU/SDC System for analysis. Each liquid sampler utilizes a scintillation detector to sense for intersystem radiation leakage. A high radiation level indicates leakage of reactor coolant into the RCCWS System. The high level will be alarmed in the main control room.~~

Intersystem leakage of radioactive material into each RCCWS train is monitored continuously by the PRMS. An in-line radiation monitor is provided at the RCCWS common discharge line that connects the cooling water output flows from the RWCU/SDC non-regenerative heat exchanger, the FAPCS heat exchanger, the upper and lower drywell coolers, the reactor building chiller, and the RCCWS air cooler. A high level of radioactivity is indicative of reactor coolant leakage into the closed loop RCCWS train. The high radiation level will be alarmed in the control room.

Differential Temperature Monitoring in Equipment Areas

Differential temperature monitoring is provided in key areas in the reactor building to detect for small leaks. Such areas as the main steamline tunnel and the equipment areas

RAI Number: SPLB.24

Question:

SSAR Section 5.2.5 Reactor Coolant Pressure Boundary

RG 1.45 Position C.7 states that procedures for converting various indication to a common leakage equivalent should be available to the operators. Explain how SBWR will comply with this position.

GE Response:

SBWR SSAR Section 5.2.5.8, Position C.7 subparagraph will be revised in Amendment 1 of the SSAR (see attached) as follows:

"Each monitored leakage parameter is indicated in the main control room and will activate an alarm on abnormal indication. Procedures will be provided to the operator to convert the identified and unidentified leakages into a common leakage rate equivalent to determine that the total leakage rate is within the technical specification limit. Each monitored leakage channel of LD&IS can be tested and calibrated separately during normal plant operation without causing a plant outage. This information satisfies RG 1.45, Position C.7."

5.2.5.6 Separation of Identified and Unidentified Leakages in the Containment

Identified and unidentified leakages from sources within the drywell are collected and directed to separate sumps, the LCW equipment drain sumps for identified leakages and the HCW floor drain sumps for unidentified leakages.

5.2.5.7 Testing, Calibration and Inspection Requirements

The requirements for testing, calibration and inspection of the LD&IS are covered in Subsection 7.3.3.4.

5.2.5.8 Regulatory Guide 1.45 Compliance

This regulatory guide specifies acceptable leak detection methods and flow rate limits for use in monitoring and detecting leaks from the reactor coolant pressure boundary.

Leakage is collected separately in drain sumps from identified and unidentified sources in the containment and total flow rate from each sump is independently monitored, thus satisfying Regulatory Guide 1.45, Position C.1.

Leakage from unidentified sources from inside the drywell is collected into the floor drain sump and monitored with an accuracy of 3.8 liters/min (1 gpm), thus satisfying Regulatory Guide 1.45, Position C.2.

There are three separate detection methods are used for leakage monitoring: (1) the floor drain sump level and pump operating frequency, (2) radioactivity of the airborne particulates, and (3) the drywell air coolers condensate flow rate, thus satisfying Regulatory Guide 1.45, Position C.3.

Intersystem radiation leakage into the Reactor Component Cooling Water System is monitored as described in Subsection 5.2.5.2.2, thus satisfying Regulatory Guide 1.45, Position C.4.

The monitoring instrumentation of the drywell floor drain sump, the air particulates radioactivity, and the drywell air cooler condensate flow rate are designed to detect leakage rates of 3.8 liters/min (1 gpm) within one hour, thus satisfying Regulatory Guide 1.45, Position C.5.

The leak detection system required to perform isolation functions is classified Class 1E, Seismic Category I; and the system is designed to operate during and following seismic events. The airborne particulate radioactivity monitor is designed to operate during an SSE event. Thus, Regulatory Guide 1.45, Position C6 is satisfied.

~~The appropriate monitored leak detection parameters and alarms are provided in the main control room. This satisfies Regulatory Guide 1.45, Position C.7. The LD&IS~~

~~instrumentation is normally tested and calibrated during normal plant operation to verify operability.~~

Each monitored leakage parameter is indicated in the main control room and will activate an alarm on abnormal indication. Procedures will be provided to the operator to convert the identified and unidentified leakages into a common leakage rate equivalent to determine that the total leakage rate is within the technical specification limit. Each monitored leakage channel of LD&IS can be tested and calibrated separately during normal plant operation without causing a plant outage. This information satisfies RG 1.45, Position C.7.

The LD&IS sensors and channels are periodically tested and calibrated during reactor operation, thus satisfying Regulatory Guide 1.45, Position C.8.

The following methods are used to verify operability:

- simulation of signals to initiate trips;
- channel-to-channel comparison of the same monitored leakage parameter;
- operability checks by comparing one method with another; and
- continuous monitoring of leakage parameters.

The limits established for alarming unidentified and identified leakages are 19 liters/min (5 gpm) and 95 liters/min (25 gpm), respectively. This satisfies Position C.9 of Regulatory Guide 1.45.

5.2.6 COL License Information

Overpressure Protection

The COL applicant is required to submit an overpressure protection analysis for core loadings different than the reference SBWR core loading.

5.2.7 References

5.2-1 "Guideline for Permanent BWR Hydrogen Water Chemistry Installations: 1987 Revision", EPRI NP-5203-SR-A.

5.2-2 B.M. Gordon, "Corrosion and Corrosion Control in BWRs", NEDE-30637, December 1984.

5.2-3 B.M. Gordon et al, "Hydrogen Water Chemistry for BWRs — Materials Behavior", EPRI NP-5080, Palo Alto, CA, March 1987.

RAI Number: SPLB.25

Question:

SSAR Section 6.7, Main Steam Isolation Valve Leakage Control System, states that the SBWR alternate to a main steam isolation valve leakage control system is contained in Appendix 19H. The staff finds that Appendix 19H to the SSAR, entitled as USI/GSI Applicability, does not contain any information, and the applicant indicates that the information will be provided by February 28, 1993. The staff cannot start review of this subject until the promised information is provided, and the review schedule should be developed based on the revised schedule of the submittal.

The staff has reviewed the information on the same subject in the EPRI Requirements Document for Passive Plants. It was identified as an open issue, page 1B.O-6 in the staff's evaluation documented in the draft safety evaluation report (DSER), Section 2.3.1 and Item II.E of Annex A of Appendix of Chapter 1. The applicant's submittal should address the staff concern identified in the above DSER.

GE Response:

Section 6.7 will be revised in Amendment 1 (see attached) to delete reference to generic issue C-8 and Appendix 19H.

Specific requirements for MSIV leakage path integrity during and following an SSE will be added to Sections 3.2 and 10.4 in Amendment 1 of the SSAR, as identified in response to ECGB.3. These additional seismic analyses and quality group requirements will be consistent with the ABWR approach, based on EPRI Evolutionary Plant SER Section 2.3.1.

6.7 Main Steam Isolation Valve Leakage Control System (BWR)

The SBWR does not have ~~alternate to~~ a main steam isolation valve leakage control system ~~is contained in the proposed resolution to generic issue C-8, Main Steam Line Valve Leakage Control Systems contained in Appendix 19H~~. The ~~alternate requires that the~~ main steam piping, bypass lines, and the condenser retain their pressure and structural integrity during and following a safe shutdown earthquake (SSE). In this manner, fission products that leak past the closed MSIVs can be plated out on the large surface volume of the main steam piping plus the condenser hotwell which will minimize release to the external environment.

RAI Number: SPLB.27

Question:

SSAR Section 6.4 Control Room Area Ventilation System

SSAR Figure 21.6.4-1, Emergency Breathing Air P&ID, indicates that (1) the Pressure and Integrity of Nuclear Components (SPEC) and (2) Emergency Breathing Air System P&ID data will be provided "later." The above information is needed for the staff's compliance review of the emergency breathing air system (EBAS).

GE Response:

The additional information for the Emergency Breathing Air System (EBAS) will be provided in Amendment 1 of the SSAR.

RAI Number: SPLB.28

Question:

Provide detailed specific conformance review for each of the HVAC subsystems under SBWR SSAR Section 6.4 and Subsections 9.4.1-9.4.4 and 9.4.6-9.4.8 against the guidelines of NUREG-0800, SRP, SRP Sections 3.4.1 for flood protection, SRP Section 3.5.1.1 for protection against internally generated missiles, SRP Section 3.5.2 for protection against extremely generated missiles and SRP Section 3.6.1 for protection against high- and moderate-energy pipe breaks. This review should be in detailed fashion for each involved system and its components versus cross-referenced generalized conformance in SBWR SSAR Section 3.0.

GE Response:

The specific conformance review of each of the HVAC subsystems listed under SSAR Section 6.4 and Subsections 9.4.1 through 9.4.4 and 9.4.6 through 9.4.8 against the guidelines of NUREG-0800, SRP Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 will not be required for the following reasons:

> As noted in SSAR Section 6.4, the safety-related HVAC for the control room consists of the Emergency Breathing Air System (EBAS), which is completely enclosed within the Sealed Emergency Operating Area. Details of the EBAS are provided in Subsection 9.4.1, and the safety design basis is provided in Subsection 6.4.1.1.
>
> The HVAC subsystems noted in SSAR Subsections 9.4.1 through 9.4.4 and 9.4.6 through 9.4.8 (with the exception of the EBAS referred to above) are all classified as non-safety related.

RAI Number: SPLB.29

Question:

SSAR Subsection 6.4.1.1 states that the "Sealed Emergency Operating Areas (SEOA)" envelope is sufficiently leak tight to maintain positive differential pressure of 34.5 Pa (0.005 psi) with the EBAS in operation. Also, SSAR Subsection 6.4.3 states that the SEOA boundary walls are designed with low leakage constructions, all boundary penetrations are sealed. SSAR Table 15.6-9 identifies unfiltered in flow of equivalent to 0.5 cubic feet per minute (cfm).

The staff considers 0.5 cfm unfiltered inleakage for the entire control room envelope unrealistic as judged from the to-date experience of the existing operating plants. Reassess the unfiltered infiltration inside SEOA envelope and provide credible infiltration inleakage which can be supported by approved methodology and which can be tested periodically and verified. Also, provide (1) the expected revised unfiltered infiltration rate in the SEOA envelope and (2) value credited for the entire SEOA envelope infiltration rate in accident dose calculations. Explain in detail how the SEOA envelope is isolated during accident conditions in order that it does not exceed the to be revised value of the unfiltered infiltration rate used in accident dose calculations. Identify the permanent measures to be implemented including sealing the SEOA envelope and periodic verification and testing provisions. If sealants are used, provide their acceptability and qualification to maintain needed isolation through the proposed design plant life.

GE Response:

The Sealed Emergency Operating Areas (SEOA) for the SBWR are designed using methods and design features that are significantly improved compare to those used on currently licensed plants. Some of the SBWR design features include double air lock door entrances with positive clean air flushing; electrical power penetrations similar to those used for containment penetrations; and instrumentation signal transmission using laser light through a permanently sealed glass barrier. The use of these advanced and conservative design features will result in net inflow leakage into the SEOA which is 0.5 cfm or less. The leakage rate of the SEOA is measured as part of the Emergency Breathing Air System (EBAS) surveillance requirement. See SR 3.7.1.2 in SSAR Section 3.7.1

RAI Number: SPLB.30

Question:

The SSAR states that onsite storage tanks will be located to allow drainage without damaging site facilities. How will this be accomplished?

GE Response:

All major, above-grade, onsite storage tanks will be located so that if catastrophic tank failure occurs, the liquid released will not flood the Reactor Building. The site contour will be controlled by use of appropriate berms, drainage ditches, and/or grading to direct water away from the Reactor Building.

RAI Number: SPLB.31

Question:

What are the probable maximum precipitation and probable maximum flood?

GE Response:

The probable maximum precipitation and probable maximum flood are site-specific parameters. The probable maximum flood (PMF), as described in ANSI/ANS 2.8, is 0.3 meter (1 foot) below grade. The probable maximum precipitation (PMP) is defined as 49.3 cm/hr (19.4 in./hr) or 15.7 cm (6.2 in.) in 5 minutes for rainfall and 2.394 kPa (50 lbf/ft^2) for snow loads. The plant grade will be established so that the Reactor Building exterior access openings will be at least 0.31 meter (1 foot) above the design flood level resulting from either condition. Site grading and drainage, as well as the Reactor Building roof drainage, will be designed to ensure that the maximum rainfall accumulation is limited to design values necessary to achieve the design discharge flowrate.

RAI Number: SRXB.1

Question:

SSAR Section 1.2.2.6 Remote Shutdown System. Traditionally controls for safety relief valves (SRVs) are given in the remote shutdown panel. But for the SBWR, no controls are provided for SRVs or automatic DPVs in the remote shutdown panel. Explain why no controls are required for SRVs or DPVs. Also, provide the basis for selection of controls and instrumentation to be included on the remote shutdown panel.

GE Response:

Normally the turbine bypass valves will automatically control reactor pressure. With this function available reactor cooldown is achieved through the normal heat sinks. This cool down process can be supplemented from the remote shutdown panel using the Reactor Water Cleanup/Shutdown System (RWCU/SDC). The RWCU/SDC system provides the capability to bring the reactor from high pressure to cold shutdown.

If main steam line isolation occurs, the Isolation Condenser system, (ICS), will automatically control reactor pressure. Since the logic processing equipment for the ICS is located in the Safety Envelope, the event necessitating control room evacuation will not impact ICS operation, and continued operation of the isolation condensers can be assumed.

The SBWR systems and equipment needed to achieve an orderly shutdown were identified. Then functional task analyses were performed for each identified system to establish the controls and indicators needed for RSS control.

RAI Number: SRXB.2

Question:

SSAR Section 1.2.2.6 Remote Shutdown System. Electrical power distribution system is included in the remote shutdown panel. But it is not clear whether diesel generators controls are provided in the remote shutdown panel.

GE Response:

If the diesel generator needs to be started the RSS has a control switch that, when actuated, disables the control of the start and stop signals from the control room. Then the diesel generator can be stopped or started by a local switch located near the diesel generator.

RAI Number: SRXB.4

Question:

SSAR Section 1.2.2.4.1. Reference to reactor water cleanup/shutdown cooling system is incorrect because it refer to the same section in the SSAR (Section 1.2.2.4.1).

GE Response:
Section 1.2.2.4.1 will be corrected to refer to Section 5.4.8 in Amendment 1 of the SSAR (see attached).

1.2.2.4 Core Cooling Systems

1.2.2.4.1 Reactor Water Cleanup/Shutdown Cooling System

See discussion in ~~Subsection 1.2.2.4.1~~Subsection 5.4.8.

1.2.2.4.2 Isolation Condenser System

The Isolation Condenser System (ICS) removes decay heat after any reactor isolation during power operations. Decay heat removal limits further pressure rises and keeps the RPV pressure below the SRV pressure setpoint. It consists of three independent loops, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC/PCC pool which is vented to the atmosphere.

The ICS is initiated automatically on either a high reactor pressure, or MSIV closure, or a Level 2 signal. To start an IC into operation, the motor-operated condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The ICS can also be initiated manually by the operator from the MCR. A pneumatic-operated condensate return bypass valve is provided for each IC which opens if the 125 Vdc power is lost.

The ICS is isolated automatically when either a high radiation level or excess flow is detected in the steam supply line or condensate return line.

The IC/PCC pool is divided into subpools which are interconnected at their lower ends to provide full use of the water inventory for heat removal by any IC. The IC/PCC pool is normally cooled by the FAPCS. During IC operation IC/PCC pool water will boil, and the steam produced will be vented to the atmosphere. This boil-off action of nonradioactive water is a safe means for removing and rejecting all reactor decay heat.

The IC/PCC pool has an installed capacity that provides at least 72 hours of reactor decay heat. The heat rejection process can be continued indefinitely by replenishing the IC/PCC pool inventory. If normal make-up systems are unavailable, make-up can be provided via post-LOCA pool water make-up connections located just above grade level outside the reactor building. These lines are classified Quality Group C and Seismic Category I. This make-up can be accomplished without any valving changes in the reactor building no matter what the prior operating mode of the FAPCS might have been.

The ICS passively removes sensible and core decay heat from the reactor (i.e., heat transfer from the IC tubes to the surrounding IC/PCC pool water is accomplished by natural convection, and no forced circulation equipment is required) when the normal heat removal system is unavailable following any of the following events:

RAI Number: SRXB.6

Question:

SSAR Section 1.7. Does this section include drawing standards, piping and instrumentation diagrams (P&ID), P&ID standard symbols, graphical symbols for use in instrument electrical diagrams (IEDs), etc., similar to the submittal given in ABWR SSAR Section 1.7?

GE Response:

GE will incorporate a drawing standard in Amendment 1 of the SBWR SSAR that will be similar to the ABWR SSAR Section 1.7.

RAI Number: SRXB.7

Question:

SSAR Section 3.1.4.4. General Design Criteria (GDC) 33 requires a system to supply reactor makeup for protection against small breaks in the reactor coolant pressure boundary. GE takes credit for automatic depressurization (ADS) and integrated control (ICS) systems even though they are not water injection systems or make-up systems to meet GDC 33. The CRD system which can be used for water injection is not a safety-grade system. GE should explain in detail why a safety grade high-pressure core injection system is not necessary to meet GDC 33.

GE Response:

10CFR50 Appendix A, Criterion 33 requires that a system be provided to make up reactor coolant that may be lost as a result of small breaks in the reactor pressure boundary. The purpose of this system is to prevent degradation of specified acceptable fuel design limits. The system must also be able to operate with the loss of either off-site or on-site power. There are no requirements for a safety-related high-pressure injection system, although Criterion 33 suggests that a safety system function that provides reactor coolant inventory control during normal operations may be used.

The SBWR meets the requirements of Criterion 33 by using the safety-related Gravity-Driven Cooling System (GDCS) to provide emergency core cooling after any event that threatens the reactor coolant inventory. The GDCS requires no external ac electrical power source or operator action. The safety-related Automatic Depressurization Subsystem (ADS), is capable of automatically and quickly depressurizing the reactor vessel, to allow the GDCS to replenish core coolant and maintain core temperatures below design limits. The ADS also requires no external ac power source or operator action. The GDCS and ADS are designed such that they may also be manually operated from the Control Room.

Although GDCS and ADS together provide the safety-related mitigation of all breaks threatening reactor coolant inventory, it is desired to minimize the probability of their actuation for small breaks of the type addressed in Criterion 33. This is accomplished in SBWR with the non-safety-related high-pressure make-up function of the Control Rod Drive (CRD) System. Make-up coolant water to the reactor is provided by the CRD System via the Reactor Water Cleanup/Shutdown Cooling System piping to one of the feedwater lines, and in turn to the core, any time feedwater flow is unavailable. This mode of operation will actuate and operate automatically upon receipt of a low reactor water level (Level 2) signal, or may be initiated manually by the operator. The flow capacity of the CRD System is sufficient to make up for the inventory lost through a break in the reactor coolant pressure boundary with an effective flow area of up to 5.067×10^{-4} m^2 (0.00545 ft^2). The system is designed so that it can perform its make-up function with or without off-site power available.

In summary, GDCS and ADS provide the ultimate safety-related mitigation of reactor coolant pressure boundary breaks. The high-pressure make-up function of the CRD

System is provided to reduce the probability of GDCS and ADS actuation for small breaks of the type specified in Criterion 33.

RAI Number: SRXB.8

Question:

SSAR Section 4.1, Summary Description, references a non-existing Subsection 1.3.1.1 for a summary of the important design and performance characteristics, some of which are given in SSAR Table 1.3-1 and Tables 4.4-1 and 4.4-2. Please provide a complete summary table as required by the SRP Section 4.1.

GE Response:

In Amendment 1 (see attached), SSAR Section 4.1 will be revised to refer to Table 1.3-1, which includes comprehensive data on reactor system design characteristics. This section follows Regulatory Guide 1.70, Revision 3; an SRP Section 4.1 does not exist. Tables 4.4-1 and 4.4-2 include additional data required for SSAR Section 4.4.

4.0 Reactor

4.1 Summary Description

The reactor assembly consists of the reactor pressure vessel, pressure containing appurtenances including control rod drive (CRD) housings and in-core instrumentation housings, plus the reactor internal components described in Subsection 4.1.2. Figure 5.3-3 (Reactor Key Features) shows the arrangement of the reactor assembly components. A summary of the important design and performance characteristics of the reactor and plant is given in ~~Subsection 1.3.1.1~~ Table 1.3-1. Loading conditions for reactor assembly components are specified in Subsection 3.9.5.2.

As explained in Section 4.2, a typical fuel and control rod design and core loading pattern adapted for SBWR is used as the basis for the system response studies in Section 6.3 and Chapter 15. The actual fuel and control rod designs and core loading pattern to be used at a plant is required to meet criteria approved by the NRC, and will be provided to the NRC for information. The typical fuel and control rod design and core loading pattern are presented in this chapter; information to be provided by the utility referencing the SBWR design is contained in the interface subsections.

4.1.1 Reactor Pressure Vessel

The reactor pressure vessel includes the shroud support brackets. Flow restrictors are included in the steam outlet nozzles and the GDCS/equalizing line nozzles. The reactor pressure vessel design and description are covered in Section 5.3.

4.1.2 Reactor Internal Components

The major reactor internal components include (1) the core (fuel, channels, control rods and instrumentation), (2) the core support structures (shroud, shroud support, top guide, core plate, and integral control rod guide tube and orificed fuel support), (3) chimney and partitions, (4) chimney head and steam separators assembly, (5) steam dryer assembly, (6) feedwater spargers, (7) standby liquid control header, sparger and piping assembly, and (8) in-core guide tubes. Except for the Zircaloy in the reactor core, these reactor internals are stress corrosion-resistant stainless steels or other high alloy steels. The fuel assemblies (including fuel rods and channels), control rods, chimney head and steam separator assembly, steam dryers and in-core instrumentation assemblies (Subsection 4.1.2.1) are removable when the reactor vessel is opened for refueling or maintenance.

4.1.2.1 Reactor Core

Important features of the reactor core (Figure 4.1-1)are:

RAI Number: SRXB.9

Question:

SSAR Subsections 4.1.4.2, Fuel Design Analysis, 4.1.4.3, Reactor System Dynamics, 4.1.4.4, Nuclear Analysis and 4.1.4.6, Thermal-Hydraulic Calculations, state that nuclear and thermal-hydraulic analysis techniques and computer codes are "based on" or "adapted" using NRC-approved criteria. Please discuss and provide additional references and/or approved code names to satisfy SRP 4.1 and 4.3.3 requirements.

GE Response:

For steady-state evaluation of the SBWR core performance, the following NRC-approved codes are used:

(1) TGBLA – infinite lattice multi-group diffusion theory code to provide group constants, infinite lattice neutron multiplication, and rod-by-rod power peaking for input into the 3D BWR simulator PANACEA. TGBLA is used to evaluate each unique lattice nuclear design in the SBWR core.

(2) PANACEA – 3D coupled nuclear and thermal-hydraulic BWR simulator. PANACEA takes input from TGBLA as well as reactor state user input and calculates core neutron multiplication and core power distributions to determine margin to minimum critical power ratio (MCPR) and maximum linear heat generation rate (MLHGR) limits and to determine the cycle energy and fuel exposure distribution. PANACEA is also used to determine cold shutdown margin and hot excess reactivity. PANACEA is also used to determine the scram reactivity and 3D void coefficient.

(3) VCOF – point model void coefficient to determine core void coefficient as a function of void fraction.

(4) CRNC – used to supply reactor state 1D neutronics parameters to the reactor/plant thermal-hydraulics transient codes.

RAI Number: SRXB.10

Question:

SSAR Section 4.2, Fuel System Design, states the fuel to be used in the SBWR is "any fuel design that is based on an NRC-approved design or meets the criteria documented in Appendix 4B." Reference is made to Amendment 15 to NEDE-24011-P-A, which applies to 8x8 and 8x8R operating reactor lattice geometries. Explain why compliance with the referenced acceptance criteria for operating reactor fuel is considered sufficient for the shorter SBWR fuel. This approach was rejected in favor of a reference fuel and core design for the ABWR. Provide a reference design or explain why the SBWR fuel and core design approach should be different from that approved for ABWR.

GE Response:

The SBWR fuel licensing approach is identical to ABWR, i.e., acceptance criteria are given and an example fuel design is provided to show that the criteria can be met. Neither the SBWR nor ABWR fuel designs described in their respective SSARs should be considered a reference design. The referenced statement will be corrected to read in Amendment 1 (see attached) of the SSAR "any fuel design that meets the criteria documented in Appendix 4B."

4.2 Fuel System Design

The fuel to be loaded in an SBWR is any fuel design that meets ~~is based on an NRC-approved design or meets~~ the criteria documented in Appendix 4B. Using these designs will assure that all fuel system design requirements are met.

To demonstrate the SBWR system response in this SSAR, a reference core (see Figure 4.3-1) is used which is based upon a current NRC approved fuel design (BP8x8R), but modified to account for the shorter active fuel length. BP8x8R fuel design information is provided in Reference 4.2-1. Each utility referencing the SBWR design may have different fuel and core designs which will be provided by the COL applicant to the NRC for information (refer to Subsection 4.2.1).

The control rods perform the dual function of power shaping and reactivity control. A discussion of the rod control drive system components is presented in Section 4.6.

The control rod design to be used in an SBWR is any design that is based on an NRC-approved design or meets the criteria documented in Appendix 4C. To demonstrate the SBWR system response in this SSAR, a reference SBWR control rod design is used which is based upon a typical BWR design of sheathed cruciform array of stainless steel tubes filled with boron-carbide but incorporates the shorter length. The SBWR control rod design is shown in Figure 4.2-1, which may alternatively use a combination of boron-carbide filled tubes and hafnium plates; the BWR design information is provided in Refrrence 4.2-2. The control rod design to be used at a plant will be provided to the NRC for information by the COL applicant (refer to Subsection 4.2.1).

4.2.1 COL License Information

Fuel Design

The fuel bundle name and a reference to documentation of the fuel design will be provided to the NRC for information by the COL applicant. (See Section 4.2.)

Control Rod Design

The control rod model and a reference to documentation of the control rod design will be provided to the NRC for information by the COL applicant. (See Section 4.2.)

RAI Number: SRXB.11

Question:

SSAR Section 4.2 states that the control rod design to be used in the SBWR is "any design that is based on an NRC-approved design or meets the criteria documented in Appendix 4C." "Compliance with these criteria constitutes NRC acceptance and approval of the designs without specific NRC review" is incorrectly stated. Please provide further justification for the use of currently approved designs for SBWR applications.

GE Response:

The SSAR, Section 4.2 will be changed in Amendment 1 as shown on the attached markup.

4.2 Fuel System Design

The fuel to be loaded in an SBWR is any fuel design that meets ~~is based on an NRC approved design or meets~~ the criteria documented in Appendix 4B. Using these designs will assure that all fuel system design requirements are met.

To demonstrate the SBWR system response in this SSAR, a reference core (see Figure 4.3-1) is used which is based upon a current NRC approved fuel design (BP8x8R), but modified to account for the shorter active fuel length. BP8x8R fuel design information is provided in Reference 4.2-1. Each utility referencing the SBWR design may have different fuel and core designs which will be provided by the COL applicant to the NRC for information (refer to Subsection 4.2.1).

The control rods perform the dual function of power shaping and reactivity control. A discussion of the rod control drive system components is presented in Section 4.6.

The control rod design to be used in an SBWR is any design that ~~is based on an NRC approved design or~~ meets the criteria documented in Appendix 4C. ~~To demonstrate the SBWR system response in this SSAR, a reference SBWR control rod design is used which is based upon a typical BWR design of sheathed cruciform array of stainless steel tubes filled with boron carbide but incorporates the shorter length. The SBWR control rod design is shown in Figure 4.2-1, which may alternatively use a combination of boron carbide filled tubes and hafnium plates; the BWR design information is provided in Reference 4.2-2.~~ The reference SBWR control rod design consists of a sheathed cruciform array of stainless steel tubes filled with boron carbide (B_4C) powder. Figure 4.2-1 is an illustration of the reference design. The main structural members of the reference design are made of stainless steel and consist of a top handle, a lower transition piece with a control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The top handle, lower transition piece, and the U-shaped sheaths are welded into a single skeletal structure. The U-shaped sheaths are welded into the center post, handle, and lower transition piece to form the housing for the absorber rods filled with B_4C. Above the handle extends four support ears which provide lateral support of the upper end of the control rod when the fuel assemblies have been removed, thereby eliminating the need for blade guides. Rollers at the top of the control rod guide the blade between the fuel channels while rollers at the sides of the coupling provide lateral supportand guide the lower end of the control rod inside the cruciform shaped control rod guide tube, as the rod is inserted and withdrawn from the core. The B_4C powder in the absorber tubes is compacted to approximately 70% of its theoretical density. The B_4C is sealed into the absorber tubes by plugs welded into each end, and is longitudinally separated into individual compartments by stainless steel balls. Typical parameters of the reference SBWR control rod design are provided in Table 4.2-1. The control rod design to be used at a plant will be provided to the NRC for information by the COL applicant (refer to Subsection 4.2.1).

4.2.2 References

4.2-1 GE Fuel Bundle Designs, NEDE-31152P.

~~4.2-2 GE Control Rod Designs, (to be issued).~~

Table 4.2-1 Typical Parameters — Reference SBWR Control Rod Design

Control Rod Weight	77 kg (170 lb)
Overall Poison Length	2591 mm (102 in)
Control Rod Drive Stroke	2616 mm (103 in)
Absorber Rod — B_4C	
Number per control rod	72
Diameter	4.22 mm (0.166 in)
Density (% theoretical)	70%
Absorber Tube — B_4C	
Cladding material	304 S.S.
Outside diameter	5.59 mm (0.220 in)
Wall thickness	0.69 mm (0.027 in)
Sheath Thickness	1.14 mm (0.045 in)
Roller Material	Inconel X-750
Roller Pin Material	PH13-8Mo

RAI Number: SRXB.12

Question:

SSAR Section 4.3 provides an "example" core loading map for a typical equilibrium-cycle core (with a currently approved fuel design (BP8x8R)) which is used for the system dynamic response analyses given in SSAR Section 6.3, Emergency Core Cooling Systems, and Chapter 15, Accident Analyses. Please provide results for a typical initial core and discuss the requirements for transition cycle analysis.

GE Response:

The details of the nuclear design do not affect the loss-of-coolant-accident (LOCA) analysis in Chapter 6.3 because the decay heat assumed bounds the core exposure in either case.

Sensitivity studies are performed to determine the effect of cycle exposure on Chapter 15 transients. SBWR does not have the flow increase transients, which are more severe at low exposure. Events which must be reanalyzed for specific core configurations (e.g., initial, transition, and equilibrium cycles) are specifically identified as such in Chapter 15. The approach used is consistent with the GESTAR approach in which a small set of events are reanalyzed for specific core configurations.

The results for a typical initial core are contained in *Core Design and Evaluation - Initial Cycle*, G.E. Doc. No. 25A5027, Class III (proprietary), which will be provided under a separate cover letter. Each operating cycle is required to satisfy the safety envelope (i.e., limits on minimum critical power ratio [MCPR], and maximum linear heat generation rate [MLHGR], cold shutdown margin, stability decay ratio, etc., must be complied with in the design and in operation). Some limits, such as MLHGR and MCPR, are explicitly determined for each cycle of operation and feed directly into the design of the loading pattern and bundle nuclear design and operating trajectory. They are also monitored by the plant process computer. Compliance with other requirements, such as stability decay ratio, can be demonstrated by showing that the important core characteristics that define the dynamic performance (i.e., void reactivity coefficient, power distribution, etc.) reside within the bounds of the inputs used in the analyses provided in the SSAR Chapter 15. Should the as-designed core not reside within the bounds of the analyses in Chapter 15 of the SSAR, then specific analyses shall be performed to demonstrate compliance.

RAI Number: SRXB.13

Question:

SSAR Section 4.3, by reference to Appendix 4A, Typical Control Rod Patterns, provides an "example" set of control rod patterns for an equilibrium core at rated power/flow and equilibrium xenon conditions along with the associated axial and radial power and exposure distributions at 15 cycle exposure steps. Please provide equivalent results for an initial core and discuss the approach used for transition cycle analyses to ensure the design or limiting power distributions remain within the design power peaking factor components given in the comparative design SSAR Table 1.3-1 of Chapter 1.

GE Response:

The results for a typical initial core are contained in *Core Design and Evaluation – Initial Cycle,* G.E. Doc. No. 25A5027, Class III (proprietary), which will be provided under a separate cover letter. See the response to SRXB.12 for the approach used in generating the transition cycle design.

RAI Number: SRXB.14

Question:

The relationship of the inferred power distributions to the monitoring instrumentation is not addressed as required by SRP 4.3.2.2. Please provide a discussion of the procedure used to develop the power distribution and provide justification for use of only four fixed axial incore detectors instead of having a detailed axial base shape from movable traversing incore probe (TIP)-like detectors.

GE Response:

1) "...procedure used to develop the power distribution."

The 3D core power distribution is determined based on the calculation by the 3D core simulator which uses plant inputs such as core flow, core thermal power, control rod positions, feedwater flow and temperature, etc. The step-by-step procedure to determine core power distribution is as follows:

a) Using plant inputs such as core thermal power, core flow, control rod positions, etc., the 3D core simulator calculates three dimension core power distribution based on the three dimension neutron diffusion theory as its calculation method.

b) The local core power distribution information measured by the fixed in-core detectors (gamma thermometers) located at four axial elevations next to the LPRMs at all LPRM assembly locations in the core is sent to the 3D core simulator (in the plant process computer). (These measured local power data are equivalent to the TIP data used in operating BWRs. The reason the axial power profile data points can be reduced from 24 or more to 4 is because of the specific diffusion theory adaptive model used. This is explained more in the following paragraphs.)

c) The 3D core simulator "adapts" the diffusion theory solution to the measured local power data so that the measured and estimated detector readings agree. This is the procedure under which the core power distribution is developed. The 3D core simulator core power calculation method is documented in GE report, NEDO-20340-3, Revisions 1 and 2.

In GE's 3D core simulator model, there are two calculational models. One model uses TIP-type axial power profile data as adaptive inputs. This is called the "TIP Adaptive" method. The other model uses the four discrete axial data points as adaptive inputs, corresponding to the four LPRM locations. This is called the "LPRM Adaptive" method. The LPRM Adaptive method is used in the SBWR application with the gamma thermometer design. That is, for each in-core instrument location, only four measured data points are provided to the 3D core simulator for core power distribution calculations.

d) The axial power profile data calculated by the 3D core simulator are then used to calibrate the LPRM detectors. This is done by comparing the measured LPRM data with the "calculated LPRM data" determined by the 3D core simulator. A gain adjustment factor is then determined for the calibration of each of the in-core LPRM detectors based on the difference of the two data values.

2) Discussion on "...justification for use of only four fixed axial in-core detectors instead of having a detailed axial base shape from TIP - like detectors."

a) The GE 3D core simulator has been approved for three dimension core power distribution calculation applications, as documented in the GE report, NEDO-20340-3, Revisions 1 and 2.

b) The GE 3D core simulator has two methods of measured data adaptation for core power distribution calculations, the TIP Adaptive method and the LPRM Adaptive method.

c) The bundle power accuracy determined by the TIP Adaptive method was documented in NEDO-20340 (Rev. 3 and later revisions). (The uncertainty factors equivalent for SBWR application based on the TIP Adaptive method is documented in Table 7A-10 of Section 7A, SBWR SSAR, Document # 25A5113, Rev A.)

d) The additional uncertainty introduced between a core power distribution calculation based on the TIP Adaptive method versus that based on the LPRM Adaptive method is determined to be 1.4%. This is resulted from a simulation study of a whole cycle exposure tracking calculation using simulated "measured" data based on SBWR core and fuel design. The procedure and conditions of this calculation are documented in Section 7A.4.4. The result is shown in Figure 7A-19 and Table 7A-10.

e) With the additional uncertainty considered and included as shown in Table 7A-10, the overall uncertainty based on LPRM Adaptive method is determined to be 6.3%. This is better than the uncertainty value based on the conventional process computer core power calculation method using TIP. For more detailed explanation, see Section 7A.4.4 and Table 7A-10 of Section 7A, SBWR SSAR, Document # 25A5113, Rev A.

(Note: In Table 7A-10, the value "7.0%" for "PC Power to Diffusion" under "Process Comp. w/Neutron TIP" is based on reload core data. The corresponding value of "6.0%" for a group of components under "3D Core Simulator w/GT (Half Core)" is based on 3D Core Simulator calculation with Gamma TIP Adaptive inputs. This "6.0%" value, added by "1.4%" for half core interpolation error and "1.5%" for GT calibration error, gives the overall RMS value of "6.3%." The above values "7.0%" and "6.0%" used in Table 7A-10 are taken from Revision 2 of NEDO 20340-3.)

RAI Number: SRXB.15

Question:

A complete set of reactivity coefficients are not presented as required by SRP 4.3.2.3. Please provide additional information to supplement the Table 1.3-1 design end-of-cycle Doppler and void coefficient values.

GE Response:

The hot operating Doppler and void reactivity coefficient results throughout the operating cycle for a typical equilibrium core are contained in the latest revision to *Core Design and Evaluation – Equilibrium Cycle,* G.E. Doc. No. 23A6891, Class III (proprietary), which will be provided under a separate cover letter. Compliance with the requirements on the cold void coefficient, moderator temperature coefficient, and prompt power coefficient will be included in Amendment 1 of the SSAR.

RAI Number: SRXB.16

Question:

Control requirements are not provided as required by SRP 4.3.2.4, except for an example of the all-rods-in and the strongest rod withdrawn K-effective values at the stated minimum cold shutdown margin condition at the limiting cycle exposure for the reference equilibrium cycle case. Please provide the additional information required by the SRP and include initial core results.

GE Response:

The results for a typical initial core are contained in *Core Design and Evaluation – Initial Cycle,* G.E. Doc. No. 25A5027, Class III (proprietary), which will be provided under a separate cover letter. The results for the reference equilibrium core are contained in *Core Design and Evaluation – Equilibrium Cycle,* G.E. Doc. No. 23A6891, Class III (proprietary), which will be provided under a separate cover letter.

RAI Number: SRXB.17

Question:

Appendix 4D - Stability Evaluation states that "the most limiting stability condition in the SBWR normal operating region is at the rated power/flow condition" and that "the SBWR is designed so that power oscillations are not possible throughout the whole operating region (including plant startup)." Stability performance during plant startup conditions is of concern because of the possibility of a wide range of power/flow/pressure/water level and subcooling conditions as well as skewing of axial and radial power distributions due to control rod withdrawal during heatup. The current reactor heatup and pressurization procedure for the Dodewaard natural circulation BWR plant startup is similar to that outlined for an SBWR; however, the actual plant designs differ significantly. Please provide further analysis to evaluate bounding ranges of plant conditions and procedures to justify the assertion that no unstable mode is expected to be encountered during SBWR startups.

GE Response:

In Appendix 4D, the decay ratios for various steady-state operating conditions in the power/flow map were listed. It is obvious that the bounding condition for the normal operation is at the maximum power and that it still meets the stability design criteria.

For transient events, two events which produce the highest power/flow ratios were considered: (1) loss of feedwater heater for power increase case and (2) loss of all feedwater flow for water-level decrease case. The decay ratios for both events are within the design criteria. The anticipated transient without scram (ATWS) events were analyzed with the 3-dimensional TRACG code and the results presented in Chapter 15, Section 8. The results show that for all the cases with various mitigation, no power oscillation was observed.

Based on the above analyses, it is concluded that SBWR is stable under normal operating conditions throughout the entire power/flow map, abnormal transient, and ATWS conditions and that the rated power/flow condition is the most limiting point in the power/flow map with a large stability margin.

The SBWR starts up at low pressure under natural circulation conditions. It has been suggested that geysering mode instability could be a concern during such startup condition. This instability may cause startup delay if it occurs, but it is not related to plant safety. It should be noted that this has never been observed at Dodewaard.

In Appendix 4D.3, it was concluded that the geysering-mode instability is not expected during SBWR startups. This conclusion was based on (1) extrapolation to SBWR of the Hitachi's geysering test data (using Freon as a coolant) from their small-scale natural circulation test loop in Japan; (2) TRACG simulation of a typical SBWR startup, which indicated no occurrence of geysering; and (3) comparison of the thermal hydraulic conditions calculated by TRACG for SBWR startup and those measured during the

February 1992 Dodewaard plant startup, which showed similarity to the SBWR, noting that no geysering has occurred during the Dodewaard startup.

Further analyses will be performed to support the conclusion that no unstable mode is expected to occur during SBWR startups. GE plans to take the following approach:

(1) Aritomi[1] has performed the small-scale experiments which show the geysering-type phenomenon. "Geysering" here refers to a condensation induced instability caused by condensation of vapor produced in the heated section in the subcooled upper plenum. TRACG analysis of the test data is performed to demonstrate the capability of the TRACG code to predict this behavior.

(2) TRACG analysis will be performed for the Dodewaard Nuclear Plant startup, to further demonstrate the capability of the TRACG code to predict plant behavior.

(3) Then, using the TRACG code, various SBWR startup trajectories are simulated to examine the possibility of occurrence of unstable mode including the geysering-type oscillation during the startup. A boundary of unstable plant conditions, if it exists, will be identified. The unstable plant conditions will be compared with the range of normal SBWR startup conditions to assess a potential for occurrence of unstable mode during SBWR startups.

TRACG analysis of some of the Aritomi's test data has been performed to predict the geysering phenomena. Results of the analysis were reported in Section 5.6, TRACG Qualification Licensing Topical Report, NEDE-32177P, February 1993. From this analysis, it was concluded that TRACG successfully calculated the geysering oscillations seen in the experiment.

TRACG analysis of SBWR startups is now under way. A range of heatup rate encountered during startup is simulated to examine a possibility of occurrence of instability. Preliminary results show that no unstable mode occurs during startup. GE plans to submit an analysis report to the NRC by September 1993.

---

[1] J. H. Chiang, M. Aritomi, R. Inoue and M. Mori, "Thermo-Hydraulics during Startup in Natural Circulation Boiling Water Reactors," NURETH-5, 9/92.

RAI Number: SRXB.18

Question:

The SSAR states that proprietary Hitachi test data available to GE indicates that the geysering mode can be avoided if core inlet subcooling is kept near zero (0 to 9°F) during plant startup up to 2 percent of rated power. Other proprietary test data, which has been referenced in open literature by Tokyo University, suggests contradictory effects from low subcooling. Please provide an evaluation of all available test data, and discuss any additional analyses and/or tests planned to resolve this issue.

GE Response:

Most of the test data from various test facilities which GE has reviewed seems to show the effect of inlet subcooling on geysering as shown in the attached Figure 1: If the data are presented in the form of "channel inlet subcooling vs channel heater power," the stable region is either a low or high subcooling region and a region of subcooling in between is the unstable region. Therefore, depending upon how the test is conducted, there is a possibility that different conclusions may be drawn from the same trend, e.g., (1) low subcooling appears to be unstable if test is done per path *a*, (2) low subcooling appears to be stable if test is done per path *b*, and (3) low and high subcooling appear to be stable if test is done per path *c*.

The validity of direct application of the small-scale test results to the SBWR situation is questionable. Therefore, as mentioned in SRXB.17, GE's approach is to bridge the test data from small-scale test loops and SBWR's case via TRACG. In other words, TRACG is qualified against some of the test data to demonstrate capability of predicting the geysering phenomena observed in the test loop and then, using the TRACG code, SBWR startup is simulated to show that no geysering is expected to be encountered during SBWR startups.

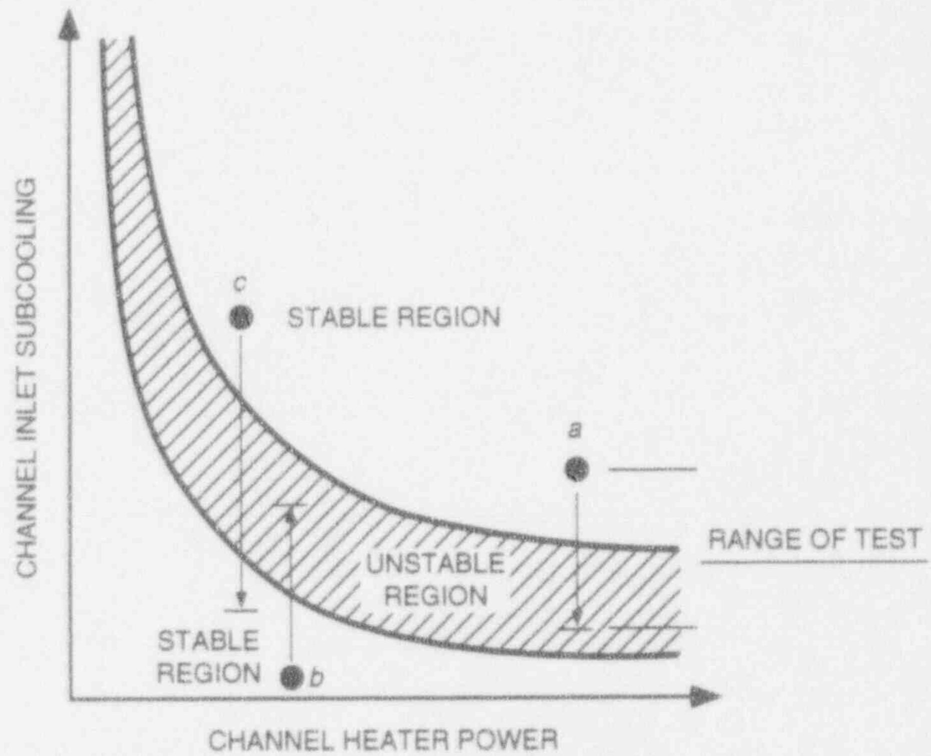


Figure 1. Effects of Inlet Subcooling

RAI Number: SRXB.19

Question:

Identify the essential portions of the CRD system which are safety related. Describe how the safety-related portions of the system are isolated from the non-essential portion of the system.

GE Response:

The following equipment constitutes the safety-related portions of the CRD system.

(a) The fine motion control rod drive (FMCRD), including the primary pressure boundary, the control rod separation switches (for rod drop accident prevention), the brake and ball check valve (for rod ejection accident prevention), the outer tube and middle flange (for drive shoot-out support), and the hollow piston latches (for post-scram control rod position holding).

(b) The hydraulic control units (HCUs) (scram circuit only).

(c) The scram insert piping from the HCUs to the FMCRDs.

(d) The HCU scram accumulator charging water header pressure instrumentation.

(e) The charging water header and purge water header air-operated isolation valves and the piping between these valves and the HCUs.

(f) The check valve, motor-operated test valve, and connecting piping at the interface of the high pressure make-up line with the RWCU/SDC system piping.

The non-safety-related portions of the CRD system interface with the safety-related portions at the following connections:

(a) The HCU charging water header supply line,

(b) The FMCRD purge water header supply line, and

(c) The scram valve air supply from the scram air header.

(d) The high pressure make-up line connection to the RWCU/SDC system.

During normal plant operation, the safety-related portions of the HCU are protected against failure in the non-safety-related portions of the charging water and purge water headers by check valves located in the HCU. During accident conditions, the HCU is isolated from the charging water and purge water supplies by an air-operated isolation valve located in each header. In addition, pressure instrumentation in the charging water header supply line provides signals to the Reactor Protection System (RPS) to cause reactor scram in the event of loss of charging water pressure. Loss of pressure in the scram air header causes the scram valves to actuate, resulting in reactor scram. This fail-safe feature is the same as provided in current BWR designs using the locking piston-type control rod drive.

The high pressure make-up line is classified safety-related (Quality Group B) at its connection to the RWCU/SDC system in order to provide interface compatibility with the safety-related (Quality Group B) RWCU/SDC piping. The motor-operated test valve and check valve in the CRD piping at this connection serve to separate the non-safety-related (Quality Group D) portions of the CRD system from the safety-related RWCU/SDC system piping.

RAI Number: SRXB.20

Question:

CRD pumps are used for high pressure make-up of the reactor. Confirm that the pumps power supply is from the diesel generator bus.

GE Response:

The control rod drive (CRD) pumps are powered from the diesel generator buses as shown on Figure 21.8.3-1, Electrical Power Distribution Single Line Diagram, and Table 8.3-1, Diesel Generator Load Table and Ratings.

RAI Number: SRXB.21

Question:

Describe the relative core location of control rods sharing a scram accumulator. Can a failure of the scram accumulator fail to insert adjacent rods? If so, discuss the consequences of that failure.

GE Response:

The assignments of the control rod drives to the HCUs and their relative core locations are shown in the attached Figure SRXB.21-1. As can be seen, the two control rods sharing the same scram accumulator are separated by several core locations. A failure of an HCU cannot result in the failure to insert adjacent control rods.

Division	HCU#	
A	1-22	44 CRs
B	23-45	45 CRs
C	46-67	44 CRs
D	68-89	44 CRs

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59						87	71	11	6	14					
55		D		69	79	84	76	21	15	3	19	7		A	
51			85	80	68	86	88	13	1	16	8	22	18		
47		74	89	72	81	70	83	2	9	4	10	5	13	2	
43		86	75	77	73	82	85	89	17	11	6	14	1	12	
39	81	70	83	71	78	74	87	72	12	7	15	3	19	17	20
35	73	82	68	76	84	79	75	77	8	20	18	16	21	9	4
31	78	88	80	69	45	28	33	23	10	5	22	47	58	66	56
27	27	32	44	39	41	43	31	55	53	57	62	54	46	60	51
23	43	40	42	26	38	30	35	50	65	52	56	49	61	48	59
19		35	24	37	29	34	40	67	63	60	51	55	53	64	
15		25	36	28	33	27	32	25	61	48	59	50	67	52	
11			41	45	31	39	24	36	66	64	46	58	63		
7		B		30	42	26	38	44	54	62	57	47		C	
3						37	29	34	49	65					

Top View

Figure SRXB.21-1 HCU to Control Rod Drive Assignments

RAI Number: SRXB.22

Question:

The CRD system in conjunction with the rod control and information system (RCIS) provides for selected control rod run in (SCRRI) to mitigate the loss of feedwater heating event. Describe in detail how the SCRRI system works.

GE Response:

The SCRRI function of the RC&IS is to insert a preselected set of control rods to their pre-established target position. This results in a reduction of core reactivity and power.

The loss of feedwater heating event is detected by two sensors in each of the feedwater lines. These sensors provide feedwater line temperature input into the Feedwater Control System (FWCS) as discussed in SSAR Section 15.1.1 and shown in Figure 21.7.7-3. The FWCS compares the feedwater temperature of each feedwater line with a setpoint. The setpoint is a function of reactor power level, which is provided to the FWCS by the APRM signal of the NMS. When the sensed feedwater temperature of either feedwater line (as shown in Figure 21.7.7-4) drops below the setpoint, each channel of the FWCS provides a signal to both channels of the RC&IS to initiate select control rod run-in. The RC&IC, after performing a two-out-of-three voting on these signals as shown in Figure 21.7.7-1, completes the SCRRI function by controlling the preselected Fine Motion Control Rod Drive (FMCRD) motors such that the control rods are driven to their SCRRI pre-established target position.

RAI Number: SRXB.23

Question:

Submit detailed drawings of the hydraulic control unit (HCU) and describe in detail the design of the HCUs.

GE Response:

The detailed HCU drawings have not been developed yet; however, the attached Figure SRXB.23-1 shows the basic HCU configuration and its constituent parts. In the following description, the HCU part numbers are given in parentheses.

Each HCU has a large nitrogen gas bottle (128) providing stored pressure to the underside of the piston in an accumulator (125) filled with high pressure water. This stored energy is held in check by the air-operated scram inlet valve (126). When a full scram condition is present, the Reactor Protection System (RPS) deenergizes the dual solenoid scram pilot valve (139), which was previously routing air to the diaphragm of the scram inlet valve to hold it closed against spring pressure. When both solenoids are deenergized, the pilot valve is repositioned to vent the air from the scram valve diaphragm, causing the scram valve to open, releasing the stored energy to the two connected FMCRDs.

Purge flow is provided to each FMCRD in order to keep the drive clean. Each HCU has a purge water manifold (147) connected by the purge water riser to the line from the CRD hydraulic system purge water header. The purge flow normally passes through a filter (145) and restricting orifice (144) mounted in the manifold. The restricting orifice controls the purge water flow rate during normal operation. The manifold also provides a flow path with a solenoid valve (143) and restricting orifice (142) that bypasses the normal restricting orifice. When one of the FMCRDs assigned to the HCU is being inserted, the RC&IS will open the solenoid valve to increase the purge flow rate. The increased flow is provided to make up for the volumetric change inside the drive as the hollow piston is inserted.

Hand-operated gate valves are provided on the scram risers (valves 101 and 140), the accumulator charging water riser (valve 113) and the purge water riser (valve 104). A hand-operated ball valve (116) is provided on the air riser to the pilot valve. These valves are normally open and are used to isolate portions of the HCU when maintenance is required.

The nitrogen instrumentation unit (148) contains a water level switch (129), a pressure switch (130), a pressure gage (131), a rupture disk (132) and a cartridge-type isolation valve (111). Leakage past the accumulator piston seals will cause the water level switch to close, activating a control room alarm. The pressure switch actuates when nitrogen pressure is low, causing an alarm to sound in the control room. The pressure gage provides a local indication of nitrogen pressure, and is used when charging the gas bottle. The cartridge valve isolates the instrumentation unit from the nitrogen bottle for maintenance. The rupture disk cartridge is provided for overpressure protection.

A check valve (115) is provided in the charging water riser to maintain accumulator pressure in the event of the loss of pressure in the CRD hydraulic system charging water header. Another check valve (138) is provided in the purge water riser to prevent the high pressure scram water in the accumulator from flowing back into purge water header when the scram valve is opened.

The accumulator can be drained for maintenance using the angle-type drain valve (107).

A quick disconnect fitting (146) and gate valve (141) are provided for connection of the HCU to portable test stations during plant shutdowns. The functions of the test stations are (1) to vent the scram insert line in a controlled manner to test the operability of the FMCRD ball check valve, and (2) to pressurize the underside of the hollow piston to cause drifting of the piston and its connected control rod into the core to measure driveline friction.

RAI Number: SRXB.25

Question:

In operating BWRs, the ball check valves ensures rod insertion in the event the accumulator is not charged or the inlet scram valve fails to open if the reactor pressure is above 600 psig. For ABWR this feature is not provided. Confirm whether this feature exists for the SBWR.

GE Response:

The SBWR fine motion control rod drive (FMCRD) design, like the ABWR FMCRD, does not have the capability of the locking piston control rod drive design to insert hydraulically using reactor pressure in the event of a failure in the hydraulic control unit (e.g., accumulator not charged or scram valve fails). However, the FMCRD has a diverse means of inserting the control rod using electric motor-driven run-in if hydraulic insertion fails. This feature provides the FMCRD with the capability to insert the control rod over the entire range of reactor operating pressures.

RAI Number: SRXB.26

Question:

The performance of essentially all types of safety/relief valves has been less than expected for a safety component. Because of reportable events involving malfunctions of these valves on operating reactors, the staff is of the opinion that significantly better safety/relief valves performance should be required of new plants. Provide a detailed description of improvements between SBWR SRVs and presently operating plants in the areas listed below. In addition, explain why the noted differences will provide the needed improvements.

i) Setpoint drift and "weeping" are generic problems. How will the SBWR SRVs resolve the generic problems.

ii) Valve and valve operator type and/or design. Include in the discussion of improvements in the air actuator especially materials used for components such as diaphragms and seals. Discuss the safety margins and confidence levels associated with the air accumulator design. Discuss the capability of the operator to detect low pressure in the accumulator.

iii) Specifications. What new provisions have been employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (especially temperature, humidity, and vibration)?

iv) Testing. Prior to installation, SRVs should be proof tested under environmental conditions and for time period representative of the most severe operating conditions to which they may be subjected.

v) Quality Assurance. What new programs have been instituted to assure that valves are manufactured to specifications and will operate to specifications?

For example, what tests are performed by the applicant to assure that the blowdown capacity is correct?

vi) Valve operability. Provide a summary of the surveillance program to be used to monitor the performance of the SRVs.

vii) Valve inspection and overhaul. Operating experience has shown that SRVs failure may be caused by exceeding the manufacturer's recommended service life for the internals of the SRV or air actuator. At what frequency do you intend to require visual inspection and overhaul of the SRVs? For both safety relief and ADS, what provisions exist to ensure that valve inspection and overhaul are in accordance with the manufacturer's recommendations and that the design service life would not be exceeded for any component of the SRV?

GE Response:

Responses to specific comments of the RAI are as follows:

i) **Comment:** Setpoint drift and "weeping" are generic problems. How will the SBWR SRVs resolve the generic problems?

Response: Significant setpoint drift in BWR safety relief valves (SRVs) is primarily a problem with certain pilot operated SRVs. Section 19H.2.23 of the SBWR SSAR addresses Unresolved Safety Issue (USI)/Generic Safety Issue (GSI) B-55, "Improved Reliability of Target Rock Safety Relief Valve." Direct acting SRVs, which are planned for the SBWR, have shown minor drift problems. A ±1% tolerance on setpoint is typically applied to the initial setting of an SRV. Increased variation from the nominal setpoint would be expected following months of service in the somewhat harsh steam environment. The latest surveillance test recommendations provided by ANSI/ASME-OM-1, which were incorporated into Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code, accommodate this variation in setpoint with a ±3% inservice tolerance. The increased tolerance for inservice setpoint will make setpoint drift less problematic.

The frequency of leakage in SRVs may be reduced by providing increased "simmer margins" – the difference between a valve's opening setpoint pressure and the normal operating pressure. The SBWR SRV setpoints will be chosen to provide the optimum simmer margin within the constraints imposed by the requirements of overpressure protection. For the SBWR, this simmer margin is maximized by using the highest set pressure allowed by ASME Code, which is expected to greatly reduce the frequency of leakage.

The SBWR spring set pressures were set high to permit the Isolation Condenser System time to initiate and turn the pressure transient around without lifting the valve. For design basis transients, the SRVs will not open because of the lower ratio of power to steam volume in SBWR. (See SBWR SSAR, Chapter 15). This means that there will be essentially no challenges to open the SRVs.

The SBWR does not have automatic power-actuated pressure relif (i.e., automatic opening of the SRVs using the SRV actuator upon receipt of a high reactor pressure vessel [RPV] steam dome pressure signal).

Most leakage in direct acting BWR SRVs has been attributed to manual actuation of the valves during preoperational testing at low inlet pressures – as low as 200 psi. Utilities that have worked with GE to perform valve preoperational testing at near operational pressure (>800 psi) have had only minor leakage problems. The SBWR plants will benefit from the lessons learned in the development of preoperational test procedures for currently operating plants.

ii) **Comment:** Valve and valve operator type and/or design. Include in the discussion of improvements in the air actuator, especially materials used for components such as diaphragms and seals. Discuss the safety margins and confidence levels associated with the air accumulator design. Discuss the capability of the operator to detect low pressure in the accumulator.

Response: Direct acting SRVs, as currently planned for the SBWR, will utilize the experience gained in qualification of SRVs for earlier generation BWRs. The actual actuator configuration will be proposed by the valve manufacturer. An electropneumatic actuator with solenoid pilot valves and air cylinder as used on the BWR-6 and ABWR SRVs is likely to be used on the SBWR. This actuator has shown good reliability and environmental capability. Materials found to have high radiation resistance combined with good thermal resistance, including specially formulated compounds, will be used consistent with the environmental requirements.

The pressure in the pneumatic supply line to the SRV Automatic Depressurization System (ADS) accumulator is monitored. Should the pressure in the pneumatic supply line to the ADS accumulator fall below the pressure required to actuate (i.e., open) the SRVs, the SRVs will be declared inoperable.

The SBWR Technical Specifications, e.g., SSAR, Chapter 16, Section 3.5, requires the plant to be Mode 3 (hot shutdown) within 12 hours after declaring three or more SRVs inoperable.

iii) **Comment:** Specifications. What new provisions have been employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (especially temperature, humidity, and vibration)?

Response: Specifications will require qualification of the valve and actuator per the latest approved regulatory and industry codes such as NUREG 0588, IEEE-382, IEEE-344, and IEEE-323 supplemented with detailed SBWR plant specific environmental conditions and requirements. These codes and regulations are updated periodically to provide new and/or improved guidelines for equipment qualification.

iv) **Comment:** Testing. Prior to installation, SRVs should be proof tested under environmental conditions and for time period representative of the most severe operating conditions to which they may be subjected.

Response: Specifications will require proof testing of the design as considered essential for demonstration that the valve design meets specification requirements, including normal, abnormal, dynamic, and DBE conditions.

v) **Comment:** Quality Assurance. What new programs have been instituted to ensure that valves are manufactured to specifications and will operate to specifications?

For example, what tests are performed by the applicant to ensure that the blowdown capacity is correct?

Response: The specification will require each individual valve assembly to be thoroughly examined and production steam tested to ensure it meets specification requirements including latest ASME Code, National Board, and SBWR specific operability requirements including leakage, setpoint, timing, and blowdown (reseat). Direct acting valves are full flow tested to ensure that the valve's performance (blowdown in particular) reflects actual inservice behavior. Flow capacity is ensured through the requirements of the National Board.

vi) **Comment:** Valve operability. Provide a summary of the surveillance program to be used to monitor the performance of the SRVs.

Response: The surveillance program is the responsibility of the plant owner. It is typically performed in accordance with ASME B&PV Code, Section XI, guidelines. The ASME Code is periodically updated to provide additional and/or improved guidelines.

vii) **Comment:** Valve inspection and overhaul. Operating experience has shown that SRVs failure may be caused by exceeding the manufacturer's recommended service life for the internals of the SRV or air actuator. At what frequency do you intend to require visual inspection and overhaul of the SRVs? For both safety relief and ADS, what provisions exist to ensure that valve inspection and overhaul are in accordance with the manufacturer's recommendations and that the design service life would not be exceeded for any component of the SRV?

Response: Valves for nuclear service are typically designed to provide at least 5 years service between required inspections and/or overhauls to coincide with ASME Section XI minimum requirements for surveillance programs. As an absolute minimum, the inspection interval must be at least one plant cycle. The valve manufacturer provides actual service requirements in the valve instruction manual, including recommended examinations, overhauls, part replacements, and testing. It is the responsibility of the plant owner to have the valves inspected, serviced, and overhauled in accordance with their surveillance program and to ensure that the surveillance program accommodates manufacturer recommendations.

RAI Number: SRXB.27

Question:

Can the safety/relief valves be closed by operators when these valves are actuated as part of the ADS function? If so, how long after ADS actuation can this be accomplished?

GE Response:

The Simplified Boiling Water Reactor (SBWR) Automatic Depressurization System (ADS) is designed such that, should the ADS initiate the Safety/Relief Valves (SRVs), the ADS logic must be manually reset before the SRVs can be closed. To reset the ADS logic, the Reactor Pressure Vessel (RPV) low water level ADS initiation signal must no longer exist.

It should be noted that, should the RPV low water level 1 ADS initiation signal re-occur, the ADS logic would re-initiate, and should the necessary permissive signals be present, the ADS logic would re-open the SRVs.

For long term SBWR post-accident scenarios, the RPV water level stabilizes about 1 meter above the Top of Active Fuel (TAF). This is below the RPV low water level ADS initiation setpoint.

RAI Number: SRXB.28

Question:

The isolation condenser system (ICS) is a safety-related system and the SBWR ICS design is similar to the ICS in operating plants like Dresden 2 and 3, Millstone 1, Nine Mile Point 1 and Oyster Creek. But the operational experience with the ICS in those plants has been of concern to the staff. The staff's experience with operational events relating to the ICS has indicated numerous design deficiencies and several operational problems. Has GE performed a systematic study of the operational experience related to ICS plants? What design changes and improvements have been made to the SBWR ICS design to correct potential design deficiencies in operating ICS plants?

GE Revised Response:

The ICS has proven itself as a reliable reactor decay heat removal system. Operating plant critical path unavailability caused by the ICS has been 0.5%, the major part of which was due to problems with sensitized stainless steel piping at one plant. Of the total plant critical path unavailability, piping contributed to almost 0.4%, whereas 0.1% is attributable to other causes.

This reliability information is based on the results of a search of two data bases that contain Isolation Condenser data. The data bases were GE's Comprehensive Performance Analysis and Statistics System (COMPASS) data, and INPO's NPRDS data. The searches were for operating BWR data relating to failures of Isolation Condenser systems and components.

The following design improvements have been included in the SBWR ICS design to improve further (reduce) the plant critical unavailability due to the ICS:

1. Use of carbon steel supply piping, Inconel tubes with butt-welded end attachments, and low carbon or nuclear grade stainless steel condensate return piping which is resistant to IGSCC.

2. The condensate return lines are continuously sloped downward from the IC to an elevation below reactor water level to avoid the trapping and collapse of steam in the drain piping.

3. The water quality of the makeup to the IC pools is such that pool boiloff to atmosphere and the surroundings should not require cleanup.

4. Three IC loops are provided, either of which will allow reactor operation at 80% of full power, and two or more IC loops will allow reactor operation at 100% or higher power. This enables plant availability goals to be met should IC valve open-close cycling problems develop during periodic operational readiness testing which is done during reactor power operation.

RAI Number: SRXB.29

Question:

The "safety-grade" isolation condenser (IC) calculations assume that the IC pool is saturated during system operation. While this minimizes the temperature difference between the primary coolant and the IC pool, it may not minimize the overall heat transfer, due to the high efficiency of heat transfer during boiling on the outside of the tubes. Show that the assumptions made result in the minimum heat transfer. If the IC pool is colder and does not boil, can heat removal still be adequately maintained?

GE Response:

Performance tests on the operating plants cited in the previous question (SRXB.28) show IC capacity to be from 180% to 200% in excess of design capacity, which GE surmises to be the result of using a fouling factor on the inside and outside of the tubes for design, whereas such fouling factors may not apply. The fouling factor for the inside of the tubes has been eliminated in the SBWR IC design sizing to re-dress this large overcapacity; even so, the IC may test out to have 140% capacity because of little or no fouling inside or outside the tubes which will allow subcooled nucleate boiling to occur on the outside of the tubes when the isolation condenser/passive containment cooling (IC/PCC) pool is below saturation temperature.

The IC will also be performance tested during initial reactor startup and power operation tests to establish its performance adequacy. Performance testing to confirm adequacy will also be done periodically during commercial operation as is now done on operating reactors.

RAI Number: SRXB.30

Question:

It has been stated that safety analyses "do not take credit" for the isolation condenser. While ignoring the presence of the IC as a heat sink may be conservative, it must still be recognized that the component is there and is in communication with the primary system. For instance, the presence of the DPVs on the IC stub lines imply that water can be drawn back through the ICS from the cold side of the primary system when the DPVs are actuated. In addition, since the IC pools are in communication with the PCCS heat exchanger (HX) pools, any pool heatup caused by IC operation will affect the operation of the PCCS. Show that there are no system interactions involving the IC that can degrade the plant response during a LOCA.

GE Response:

The IC branches off at a right angle from an 18-inch size pipe stub that directly leads to a 7-inch diameter flow restricting section of the DPV. The differential pressure caused by flow past the IC branch connection in the 18-inch pipe to the 7-inch diameter DPV opening is insufficient to cause backflow of liquid upward from the reactor vessel, through the isolation condenser, and back into the steam supply line.

The steam separation and venting system of the IC/PCC pool is sized for atmospheric boiloff of 126 MW of steam with three ICs in operation at 140% of rated flow. The maximum boiloff of steam by the PCCs is 30 MW. When the PCCs are operational, the temperature difference for heat transfer is about 80°F whereas the ICs are sized for a difference of about 340°F; therefore, the maximum pool boiloff rate with PCCs and ICs operating together at 80°F temperature difference is much less than the 126 MW pool boiloff capacity of the IC at 340° difference. The ICs and PCCs are located in separate subcompartments of the IC/PCC pool so direct hydrodynamic effects of the IC and PCC are isolated from each other.

RAI Number: SRXB.31

Question:

The staff is aware of plans to perform tests of a full-scale IC module in the "PANTHERS" test facility at SIET, Italy. These tests have been determined by the staff to be required as part of the design certification testing. No reference to any of these tests is contained in the SBWR SSAR.

The provisions of 10 CFR 52.47(b)(2) require that the specific testing supporting the certification of the design must be described as part of the application. Furthermore, SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," requires that the passive plant vendors submit their test program plans, test matrices, and, upon test completion, the qualified raw data to the NRC for review as part of the design certification process. Please provide detailed information on the IC tests at "PANTHERS," as indicated, and discuss how the data will be used to support analysis of IC performance in the SBWR.

GE Response:

References:

1. Letter PP Stancavage (GE) to RC Pierson (NRC), "Information Requested during the December 17, 1992 Meeting on the SBWR Testing Program," January 11, 1993, MFN No. 005-93, Docket STN 52-004.

2. Letter PW Marriott (GE) to R Borchardt (NRC), "Testing Program Supplement to the Simplified Boiling Water reactor (SBWR) Application for Design Certification," May 7, 1993, MFN No. 071-93, Docket STN 52-004.

3. Letter DM Crutchfield (NRC) to PW Marriott (GE), "Acceptance of the GE Nuclear Energy's (GE's) application for the Simplified Boiling Water Reactor (SBWR) Design," May 27, 1993, Docket No. 52-004.

Since the time of this question, GE has had a series of letters with the NRC on the PANTHERS test program. In References 1 and 2, GE supplied the NRC with information on how the SBWR testing programs, including PANTHERS, support the design certification. Included in these references were the test program plans and matrices or references to other submitted documents where that information is located. With the submittal of Reference 2, the NRC has accepted the application for design certification (Reference 3).

The IC for the SBWR is a vertical tube heat exchanger with vents (normally closed) going to the suppression pool. Since the design of the condenser unit is different from existing units, which have horizontal tubes, a prototype condenser will be built and tested as part of the PANTHERS Test Program. Tests on a full-size IC module will look at the thermal hydraulic performance of the unit, as well as the structural performance to ensure that the condenser will meet the 60-year life of the SBWR. The tests will be conducted in Piacenza, Italy by SIET and are scheduled to begin in late 1994.

GE does not consider the completion of these tests as necessary for SBWR certification and takes exception to the statement that they are required for certification. These component tests will simply confirm that the selected IC design will satisfy the SBWR performance requirements and provide data to quantify the margin above those requirements.

GE requests that the NRC confirm that the IC tests at PANTHERS are not necessary for design certification.

RAI Number: SRXB.32

Question:

One function of the reactor water cleanup system is to prevent thermal stratification in the reactor vessel lower head. If the system stops functioning, a stratified layer of cold water may begin to build up in the vessel lower head. This could have the effect of lowering the overall driving head for natural circulation in the primary system.

a) How will stratification affect normal natural circulation flow in the reactor vessel?

b) What impact would the stratification have on the operation of the safety systems, including the ECCS and isolation condensers, in the event of a transient or an accident?

GE Response:

Thermal stratification has always been a matter to receive engineering attention for BWRs because of the cold CRD purge flow which enters through the drives. Without recirculation pumps in operation, the bottom drain line temperature is monitored and compared to the steam dome temperature. If the delta temperature between these two measurements exceeds 145°F, restart of either pump is prohibited. This prevents rapid sweep out of the cooler layer of water at the bottom of the vessel and the resulting thermal transient experienced by the stubtubes and welds located at the bottom of the vessel.

The SBWR will be operated in the same manner as past BWR designs. The differential temperature between the dome and bottom drain will be monitored and used to control various operations which may create removal of any cold bottom head layer of water.

There are three scenarios which have the potential for causing thermal stratification at the bottom of the vessel.

Normal startup - As the cold FMCRD purge water is injected into the drives it exits at the bottom of the guide tubes and fills the guide tubes with cold water. The guide tubes then act as heat exchangers cooling the lower plenum water which settles to the bottom. Before core flow increases to about 10%, little sweeping and mixing of this cold water takes place. Because of this potential the vessel was designed with two drain lines to remove enough water to prevent stratification. The two drain lines will remove about 400 gpm of bottom head water from the plenum. The vessel water will be warmed up to about 115°F for the system leak test and to 176°F prior to pulling control rods. The water level will be above the first pickoff point of the steam separators for the warming process to enable good mixing of the vessel inventory.

At a very low power level of about 1% to 2%, the core flow will be at about 20% because of the flat shape of the power-flow map for the SBWR's natural circulation. Judging from studies and measurements at other reactors including natural circulation reactors, 20% core flow should be adequate to sweep any cooled lower plenum water up into the core even without the benefit of drain line flow removing water from the bottom of the vessel.

Hot Standby - The reactor will be at rated conditions or less (approximately 540°F) with no steam flow or feedwater flow. Decay heat should be rapidly decreasing following the reactor scram. In this situation the bottom head drain flow will provide the removal or turnover of bottom plenum water, taking any cooled water from the bottom head and putting it back into the feedwater line where it mixes with the hotter water in the annulus. The main condenser, shutdown cooling mode of RWCU or isolation condensers will be maintaining or reducing the vessel pressure.

Loss of Cleanup System - The Reactor Water Cleanup (RWCU)/Shutdown Cooling System has two totally redundant trains which can remove bottom head water; each train removes water from independent drain lines.

It is not felt that stratification will develop as a concern during either of the three conditions mentioned above. Detailed startup tests at different power levels and drain line flow rates will be performed to verify this conclusion. The plant will be operated similar to past BWRs by monitoring for the onset and the maximum analyzed levels of stratification.

A 3-D finite difference analysis was performed on the bottom plenum area, to determine the effect of minimum drain flow and very low core flows upon the capability to stratify the vessel coolant inventory. It was found that bottom drain flows as low as 5 kg/sec are sufficient to limit the temperature difference to less than 26°C (50°F).

During SBWR lead plant startup testing, the minimum core flow to prevent stratification without RWCU flow will be determined, for use in procedures which will limit operation under power/flow and RWCU flow conditions which might allow stratification.

In response to the two direct questions in SRXB.32:

a) Stratification does not effect the normal natural circulation flow in the reactor vessel because it does not occur during ordinary power generation conditions. Stratification would occur during very low power generation conditions if less than about 5 kg/sec drain flow were present. A single RWCU pump provides about twice this flow. Even if no RWCU drain flow were present to draw out the stratified layer, the depth of the layer would only be 1 to 2 meters, allowing the natural circulation flow to bypass the stratified layer with minimal pressure loss. It is the bypass of the stratified layer which allows it to persist.

b) The stratified layer occupies a relatively small dead zone at the bottom of the vessel. It does not interfere with the flow of the passive systems to or from the vessel. The GDCS and IC inject well above the lower plenum in the downcomer annulus above the core. As described above, the layer has insignificant effect on flow between the downcomer and the core.

RAI Number: SRXB.33

Question:

In the "Technical Introduction" volume provided to the staff at the September 3 presentation, the figure of the RWCU system in the "Safety and Auxiliary Systems" section shows a head spray connection. However, the slide comparing the RWCU systems in the ABWR and the SBWR in the same section indicates that the head spray does not exist in the SBWR. Which of these is correct?

GE Response:

The slide comparing the RWCU systems in the ABWR and the SBWR is correct. SBWR does not have a head spray connection.

RAI Number: SRXB.34

Question:

In a presentation to the staff on September 3, 1992, it was stated that the squib valve on the ECCS line between the suppression pool and the reactor vessel was timed to open 3 hours after an accident. In Section 6.3.2.1 of the SSAR, on page 6.3-4, it is stated that these valves are actuated 30 minutes after an accident. Which of these statements is correct?

GE Response:

The Gravity-Driven Cooling System (GDCS)equalizing lines connecting the suppression pool to the reactor pressure vessel (RPV) allow the suppression pool to provide RPV makeup after the GDCS Pools have drained into the RPV. The GDCS initiation logic for the squib valves in these lines has undergone a design change since the September 3, 1992, presentation. Now, this logic consists of a permissive signal (RPV water level <= TAF + 1.0 m) and a time delay (t >= LOCA (Level-1) + 30 minutes) combined in an AND-type logic, such that when both input conditions are TRUE, the output condition is TRUE, and the logic will cause the equalizing line squib valves to actuate.

Under the most rapidly developing LOCA water level transients within the RPV, after RPV depressurization is complete and GDCS flow injection from the elevated GDCS Pools starts, water level inside the RPV will recover to several meters above top of active fuel (TAF). This post-LOCA recovery of RPV water level peaks at a minimum of 45 minutes, and RPV water level may remain near the peak for another 3 or 4 hours. Even then, some residual coolant still remains inside the GDCS Pools. For some breaks, during this period, water losses from the RPV to the drywell and/or to the suppression pool can occur, and water level inside the RPV would then decrease. But if the GDCS equalizing line squib valves were to open while RPV water level is still well above TAF, then unnecessary blowdown of coolant via these equalizing lines could occur if the check valves leaked. (Note: During normal operation, the equalizing line check valve is not required to be leak-tight; it only has to limit reverse flows to less than control rod drive (CRD)makeup flow for the case of a squib valve inadvertently firing with the reactor at full pressure.)

The condition of ample core coverage is not threatened unless RPV water level drops well below TAF + 1.0 m. Therefore, the logic permissive signal of RPV water level <= TAF + 1.0 m to the equalizing line squib valves is desirable since it ensures adequate core coverage and prevents the possibility of blowdown of the RPV to the suppression pool.

The 30-minute time delay assures that the GDCS equalizing line squib valves will not open prior to completing the GDCS Pools draindown (i.e., will not occur within the first few minutes of the LOCA when the RPV is undergoing depressurization and RPV water level is decreasing). For some accidents the RPV water level approaches TAF (i.e., nominally 1.0 m above) due to blowdown through the break, safety relief valves, and depressurization valves. Therefore, the time delay of t >= Level-1 + 30 minutes is desirable logic input which prevents premature opening of the equalizing line squib valves.

Studies of a wide spectrum of LOCA breaks with the GDCS equalizing line squib valve logic described above show that equalizing line initiation will, in many cases, not occur until 3 to 6 hours post-LOCA. The squib valve components have already been qualified to DBA LOCA temperature and pressure conditions lasting 100 days and verify that the squib valves will still be able to actuate at the conclusion of this period. Thus, there is no adverse consequence to delaying the equalizing line squib valve opening until 30 minutes after reaching Level 1 because this time delay better conserves RPV post-LOCA coolant inventory if the check valve leaks.

RAI Number: SRXB.35

Question:

The operation of timers is crucial to the actuation of various parts of the ECCS in the SBWR. The automatic depressurization system and the gravity driven cooling system, including both GDCS pool and suppression pool injection, depend on elapsed time signals to accomplish their functions. How many timers are provided for each of these systems, and how are these timers controlled and powered? Verify that no single failure can disable all essential timing capability.

GE Response:

The Automatic Depressurization System (ADS) and Gravity Driven Cooling System (GDCS) are part of the Emergency Core Cooling System (ECCS) and are described in Sections 7.3.1.1 and 7.3.1.2 of the SSAR. The Safety System Logic and Control (SSLC) System provides the control logic processing facility that activates functions of the ECCS and other safety-related systems, and is described in Section 7.3.4.

The number of timers or timing functions for ADS and GDCS can be assessed from a review of Figure 21.7.3-1, Nuclear Boiler System LD, for ADS trip logic for DPV channels and SRV channels, and Figure 21.7.3-2, Gravity Driven Cooling System LD, for GDCS trip logic.

The timers associated with the SRVs of the ADS trip logic and the manual actuation of the ADS DPVs and the GDCS are discrete components. The timers associated with the automatic actuation of the ADS DPVs and the GDCS are microprocessor based and software controlled. No single failure can disable all essential timing capability.

RAI Number: SRXB.36

Question:

What is the water level inside the drywell during the course of a LOCA? Is there any sequence involving a break plus a single active failure that can result in an unrecoverable or unisolable loss of primary coolant outside of the containment? This includes both intersystem LOCAs and leakage of water directly from the drywell after an in-containment LOCA.

GE Response:

The drywell water level during a LOCA is dependent on the break location and specific assumptions made about external water sources such as feedwater and CRD injection. For the feedwater and CRD assumptions made in the SSAR, the water level in the drywell following GDCS injection will range from about 4 m above the basemat to about 8.5 m above RPV bottom-center, depending on break location. Over the long term, the drywell water level surrounding the RPV can continue to increase, due to suppression pool drawdown, until it reaches the elevation of intentionally placed spillover holes which connect the drywell annulus directly into the vertical LOCA vents (approximately 11 m above RPV bottom-center).

The containment pressure boundary is completely lined with steel and, consequently, is considered to be leak-proof. All piping which penetrates the containment is equipped with double isolation valves so no single failure can result in loss of coolant inventory to outside the containment. The containment isolation system is described in Section 6.2.4 of the SBWR SSAR.

RAI Number: SRXB.37

Question:

In SSAR Table 6.2-2, Generic Assumptions/Initial Conditions for LOCA Analyses, page 6.2-55, the short term analysis assumptions include use of the Moody critical flow model to calculate depressurization of the reactor vessel. It is known that the Moody model over-predicts considerably the rate of inventory loss from the vessel, and thus the rate of primary system depressurization. In conventional plants, this is generally assumed to be conservative. However, in the SBWR, the lack of a high-pressure emergency core cooling (ECC) capability makes reduction of the primary system pressure essential, and an acceleration of that rate through increased inventory and pressure reduction, permitting earlier actuation of the ADS and low-pressure GDCS, may not be conservative in calculating plant accident response. Demonstrate that the assumptions made in these calculations do indeed produce the most conservative analytical results.

GE Response:

The critical flow model utilized in the short-term LOCA analysis calculations is the Moody homogeneous equilibrium model (HEM), as noted in SSAR Table 6.2-2. In computing the containment response to LOCA, the equilibrium flow rates (i.e., vessel blowdown in the drywell) from this critical flow model are determined in terms of known stagnation properties in the vessel.

There are available a number of models to compute the maximum discharge rate of liquid-vapor mixtures from vessels resulting from a loss-of-coolant accident. Among them, there are two different Moody models: slip equilibrium model (1), and homogeneous equilibrium model (2). The slip model has been found to predict subcooled liquid and two-phase equilibrium flow rates somewhat higher than data representative of BWR blowdowns when vessel stagnation properties are used to evaluate critical flow rate. The homogeneous model has been found to agree quite well with the available data when vessel stagnation properties are used to evaluate critical flow rate. The two models are found to be in close agreement for single-phase steam flow. The fact that the homogeneous model predicts realistic inventory loss from the vessel for all cases makes its use appropriate for short-term blowdown analysis calculations.

---

(1) Moody, F.J., "Maximum Flow Rate of a Single-Component, Two-Phase Mixture," J. Heat Transfer, Trans. ASME, Ser. C, 87, 134 (1965).

(2) Moody, F.J., "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels," Non-Equilibrium Two-Phase Flows, ASME Symp. Vol., American Society of Mechanical Engineers (1975).

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

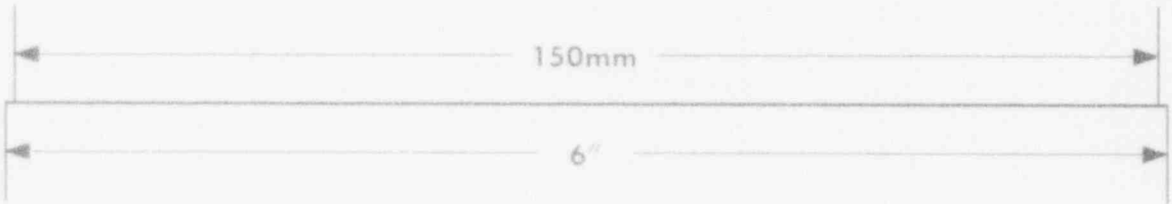
IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

RAI Number: SRXB.38

Question:

Passing reference to PCCS test programs is made in the last paragraph of SSAR Section 6.2.2.3, page 6.2-25. No specific documentation on the test programs is included in the list of references; it is inferred that the "GIRAFFE" tests at Toshiba and the planned "PANTHERS" tests at SIET are the test programs mentioned. The provisions of 10 CFR 52.47(b)(2) require that the specific testing supporting the certification of the design must be described as part of the application. Furthermore, SECY-91-273 requires that the passive plant vendors submit their test program plans, test matrices, and, upon test completion, the qualified raw data to the NRC for review as part of the design certification process. Please provide detailed information on the "GIRAFFE" and "PANTHERS" programs as indicated, and discuss how the data have been or will be used to support the assertions made regarding PCCS and containment performance. This should include any additional tests that are planned in the recently modified "GIRAFFE" loop.

GE Response:

References:

1. Letter PP Stancavage (GE) to RC Pierson (NRC), "Information Requested during the December 17, 1992 Meeting on the SBWR Testing Program," January 11, 1993, MFN No. 005-93, Docket STN 52-004.

2. Letter PW Marriott (GE) to R Borchardt (NRC), "Testing Program Supplement to the Simplified Boiling Water reactor (SBWR) Application for Design Certification," May 7, 1993, MFN No. 071-93, Docket STN 52-004.

3. JGM Andersen et al., "Licensing Topical Report, TRACG Qualification," February 1993, GE Document No. NEDE-32177P.

4. Letter DM Crutchfield (NRC) to PW Marriott (GE), "Acceptance of the GE Nuclear Energy's (GE's) application for the Simplified Boiling Water Reactor (SBWR) Design," May 27, 1993, Docket No. 52-004.

Since the time of this question, GE has [illegible]d a series of letters with the NRC on the GIRAFFE and PANTHERS test progra[illegible] In References 1 and 2, GE supplied the NRC with information on how the SBWR testing programs, including GIRAFFE and PANTHERS, support the design certification. Included in these references were the test program plans and matrices or references to other submitted documents where that information is located. With the submittal of Reference 2, the NRC has accepted the application for design certification (Reference 4).

Tests have been conducted on separate effects and integral systems effects for the PCCS at the GIRAFFE Test Facility at Toshiba, Japan. The objectives of the GIRAFFE testing program were to provide separate effects and integral systems test data for qualification of TRACG, the computer code which will be used for analysis of the SBWR containment. The separate effects tests addressed the issues of steam condensation heat transfer rates from a steam-nitrogen mixture under steady-state conditions, and of venting of noncondensable gases from PCCS to the suppression pool. Test were conducted using a full-height three-tube condenser to represent the PCC. For the venting study, the nitrogen vent line of the scaled-down heat exchanger was submerged by 0.40m, 0.65m, and 0.90m.

The integral tests demonstrated the concept of the PCCS and provide data for a variety of LOCA simulations, against which TRACG models for the containment have been qualified. The GIRAFFE Test Facility consisted of a full-scale vertical and 1:400 scale volume representation of the SBWR. Key scaled components included the RPV and containment volumes. The initial conditions for the long-term integral tests corresponded to those at one hour after LOCA occurrence. The main steam line break, GDCS line break and bottom drain line break LOCAs were simulated during the long-term system response tests. Data from these tests were used to qualify the TRACG code for SBWR applications.

GIRAFFE test data was used to support the qualification of the TRACG computer code, and this is documented in Section 5.5 of Reference 3.

At PANTHERS, a full-scale prototype PCC will be tested under simulated conditions representing a broad range of operating conditions. These tests will be conducted at the same facility in Italy as that for the PANTHERS IC tests. The major objective of these tests is to confirm, for the PCC, the thermal hydraulic performance for the SBWR service conditions. A series of tests representing a range of steam/air mixtures are scheduled for late 1993 with the test report to be issued soon after.

Following those initial performance tests, more tests will be conducted in early 1994 to gather additional thermal hydraulic performance data and structural data to qualify the design for the 60-year service life of the SBWR. GE does not consider the completion of these additional tests as necessary for SBWR certification. These component tests will confirm that the selected PCC design will satisfy the SBWR performance requirements and provide more data to quantify the margin above those requirements.

GE requests that the NRC confirm that an initial series of performance tests of the PCC at PANTHERS is all the testing necessary for design certification.

RAI Number: SRXB.39

Question:

The staff is aware of GE's plans to conduct integral long-term cooling experiments in the "PANDA" facility at the Paul Scherrer Institute (PSI). These tests have been determined by the staff to be required as part of the design certification testing. Additional tests in the "LINX" and "AIDA" facilities are also planned at PSI. No reference to any of these tests is contained in the SBWR SSAR. The provisions of 10 CFR 52.47(b)(2) require that the specific testing supporting the certification of the design must be described as part of the application. Furthermore, SECY-91-273 requires that the passive plant vendors submit their test program plans, test matrices, and, upon test completion, the qualified raw data to the NRC for review as part of the design certification process. Please provide detailed information on the "PANDA," "LINX," and "AIDA" programs as indicated, and discuss how the data will be used to support analysis of the PCCS in the SBWR.

GE Response:

References:

1. Letter PP Stancavage (GE) to RC Pierson (NRC), "Information Requested during the December 17, 1992 Meeting on the SBWR Testing Program," January 11, 1993, MFN No. 005-93, Docket STN 52-004.

2. Letter PW Marriott (GE) to R Borchardt (NRC), "Testing Program Supplement to the Simplified Boiling Water reactor (SBWR) Application for Design Certification," May 7, 1993, MFN No. 071-93, Docket STN 52-004.

3. Letter DM Crutchfield (NRC) to PW Marriott (GE), "Acceptance of the GE Nuclear Energy's (GE's) application for the Simplified Boiling Water Reactor (SBWR) Design," May 27, 1993, Docket No. 52-004.

In References 1 and 2, GE described how the PANDA confirmatory test program will be used to provide additional information of TRACG application to the SBWR. With the submittal of Reference 2, the NRC has accepted the application for design certification (Reference 3).

The Paul Scherrer Institute (PSI) of Switzerland is building an integral systems test facility (PANDA) which will demonstrate PCCS performance on a larger scale than GIRAFFE. The facility will be full-scale vertical and 1/25 scale by volume. The overall objectives of these tests are to demonstrate that the containment long term cooling performance is the same in a large scale system as previously demonstrated at a smaller scale (GIRAFFE) and that with non-uniform drywell conditions, no significant adverse effects are introduced on the performance of the PCCS.

The test series at PANDA will consist of two main steamline (MSL) break tests. The first test will duplicate the initial conditions of the GIRAFFE MSL break test with uniform drywell conditions and the second will have non-uniform conditions in the drywell. These tests will demonstrate the adequacy of the tests at GIRAFFE and are scheduled to be performed by mid 1994. These tests are not considered necessary for further TRACG qualification and certification, but are being performed to quantify the margins in the qualified TRACG code, which has been qualified using several facilities at different scales.

Data from LINX or AIDA tests are not used to support the analysis of the PCCS in the SSAR.

GE requests that the NRC confirm that the initial series of two tests at PANDA is all the testing necessary for design certification.

RAI Number: SRXB.40

Question:

The staff is aware of testing of the GDCS in the GIST facility at GE in San Jose, California. The final report on these tests has been previously made available to the staff. However, these tests are not referenced in Chapter 6 of the SSAR, nor is any indication given as to how the results have been used to support analyses of the SBWR accident response. The staff is also aware that the GDCS design represented in the GIST tests is not the same as that in the current SBWR design. The provisions of 10 CFR 52.47(b)(2) require that the specific testing supporting the certification of the design must be described as part of the application. Furthermore, SECY-91-273 requires that the passive plant vendors submit their test program plans, test matrices, and, upon test completion, the qualified raw data to the NRC for review as part of the design certification process. Please provide any detailed information on the "GIST" tests that is not included in the test program final report. In addition, discuss how the data will be used to support accident analyses for the SBWR; the discussion should include the issue of the change in GDCS design since completion of the GIST program.

GE Response:

References:

1. PF Billig, "Gravity-Driven Cooling System Integrated Systems Test - Final Report," October 1989, GE Document No. GEFR-00850, see MFN-111-92, May 13, 1992.

2. TRACG Basedeck for the GE GIST Facility, September 25, 1992, MFN 181-92.

3. JGM Andersen et al., "Licensing Topical Report, TRACG Qualification," February 1993, GE Document No. NEDE-32177P.

4. Letter PW Marriott (GE) to R Borchardt (NRC), "Testing Program Supplement to the Simplified Boiling Water reactor (SBWR) Application for Design Certification," May 7, 1993, MFN No. 071-93, Docket STN 52-004.

5. Letter DM Crutchfield (NRC) to PW Marriott (GE), "Acceptance of the GE Nuclear Energy's (GE's) application for the Simplified Boiling Water Reactor (SBWR) Design," May 27, 1993, Docket No. 52-004.

Since the time of this question, GE has had a series of letters with the NRC on the GIST test program. The GIST test program was documented in Reference 1 which includes the purpose of the program and the test matrix. Additional information including raw data has been given to the NRC in Reference 2 and other transmittals on November 2, 1992 and December 17, 1992.

The GDCS Integrated Systems Test (GIST) Facility was built at the GE Nuclear Energy site in San Jose, California. All significant plant features which could affect the performance of the GDCS (e.g., RPV, containment, depressurization system, break flows, etc.) were included in the design. GIST had a one-to-one vertical scale and a one-to-five hundred eight (1:508) horizontal area (or volume) scale of the RPV and containment volumes.

Tests run at GIST provided a qualification base for the TRACG code and also demonstrated the technical feasibility of the GDCS concept to depressurize the RPV to sufficiently low pressures to allow reflood via a gravity-fed emergency core cooling system. Four accident types were modeled at GIST; three LOCAs (main steam line, GDCS line, and bottom drain line) and a no break (isolation event) transient with loss of inventory. Data from these tests were used to qualify the TRACG code (the GE version of TRAC-BWR) for SBWR applications.

The results from GIST were used in the qualification of TRACG, which is described in Section 5.3 of Reference 3. In addition, Reference 4 documents the GIST program as a supplement to the application which has been accepted by the NRC (Reference 5).

The GIST program is complete, and there are no plans for any additional testing at GIST. GE requests that the NRC confirm that no more tests are necessary for design certification.

RAI Number: SRXB.41

Question:

A very brief discussion of the squib valve test program is made in Section 6.3.3.2, page 6.3-14, and the final test report is referenced on page 6.3-22. However, the final test report has not been made available to the staff, nor has GE indicated how the data from the test program will be used to support performance and reliability claims for the DPV valves. The provisions of 10 CFR 52.47(b)(2) require that the specific testing supporting the certification of the design must be described as part of the application. Furthermore, SECY-91-273 requires that the passive plant vendors submit their test program plans, test matrices, and, upon test completion, the qualified raw data to the NRC for review as part of the design certification process. Please provide the final test report and the additional information as indicated, and discuss how the data have been used to support the assertions made regarding DPV performance.

GE Response:

References:

1. PF Billig, "Depressurization Valve Development Test Program - Final Report," October 1990, GE Document No. GEFR-00879.

2. Letter PW Marriott (GE) to R Borchardt (NRC), "Testing Program Supplement to the Simplified Boiling Water reactor (SBWR) Application for Design Certification," May 7, 1993, MFN No. 071-93, Docket STN 52-004.

3. Letter DM Crutchfield (NRC) to PW Marriott (GE), "Acceptance of the GE Nuclear Energy's (GE's) application for the Simplified Boiling Water Reactor (SBWR) Design," May 27, 1993, Docket No. 52-004.

Since the time of this question, GE has had a series of letters with the NRC on the squib valve or DPV test program. The squib valve test program final test report (Reference 1) will be submitted to the NRC staff for closure of this RAI. In addition, Reference 2 documents the DPV development program as a supplement to the application which has been accepted by the NRC (Reference 3).

GE conducted a development test program to develop, design, build, and provide test data to qualify a DPV for the SBWR. As part of the test program, prototype valves were used in flow and reaction load tests. These tests confirmed that the DPV can be simply supported as a cantilever off the main steam line or the RPV. They also confirmed that the design will meet the flow requirements for the valve.

Environmental Qualification (EQ) dynamic loads tests were conducted to qualify the valve. These tests simulated conditions the DPV (with components aged to end-of-life) would be subjected to while sustaining plant flow and pipe induced vibrations. These tests modeled the vibrations the DPV would undergo during normal plant operation, SRV cycling, seismic events, and chugging events, and confirmed that the DPV, when called upon by an actuation signal, will perform its safety function for the SBWR.

The DPV test program is complete, and there are no plans for any additional DPV tests. The DPV represents a simple design with a high reliability to operate. The squibs have been thoroughly tested and have been shown to maintain their chemical integrity under conditions expected in the SBWR containment. GE requests that the NRC confirm that no more tests are necessary for design certification.

RAI Number: SRXB.42

Question:

The staff is aware of separate effects heat transfer tests at the Massachusetts Institute of Technology and the University of California at Berkeley, for the purpose of investigating condensation in the presence of non-condensible gases. While some information has been provided to the staff on the results of these tests, final reports have not been provided. These tests are not referenced in the SSAR, nor is any indication given as to how the results have been or will be used to support analyses of PCCS performance. The provisions of 10 CFR 52.47(b)(2) require that the specific testing supporting the certification of the design must be described as part of the application. Furthermore, SECY-91-273 requires that the passive plant vendors submit their test program plans, test matrices, and, upon test completion, the qualified raw data to the NRC for review as part of the design certification process. Please provide detailed information on the MIT and UCB tests, and discuss how the data will be used to support analyses of PCCS performance for the SBWR.

GE Response:

References:

1. Letter PP Stancavage (GE) to RC Pierson (NRC), "Information Requested during the December 17, 1992 Meeting on the SBWR Testing Program," January 11, 1993, MFN No. 005-93, Docket STN 52-004.

2. Letter PW Marriott (GE) to R Borchardt (NRC), "Testing Program Supplement to the Simplified Boiling Water reactor (SBWR) Application for Design Certification," May 7, 1993, MFN No. 071-93, Docket STN 52-004.

3. JGM Andersen et al., "Licensing Topical Report, TRACG Model Description," February 1993, GE Document No. NEDE-32176P.

4. Letter DM Crutchfield (NRC) to PW Marriott (GE), "Acceptance of the GE Nuclear Energy's (GE's) application for the Simplified Boiling Water Reactor (SBWR) Design," May 27, 1993, Docket No. 52-004.

Since the time of this question, GE has had a series of with the NRC on the heat transfer tests at the Massachusetts Institute of Technology (MIT) and the University of California at Berkeley (UCB). In References 1 and 2, GE supplied the NRC with information on how the SBWR testing programs, including separate effects heat transfer tests at MIT and UCB, support the design certification. Included in these references were the test program plans and matrices or references to other submitted documents where that information is located. The university test programs test data are used as the basis for the wall condensation model in the TRACG computer code. This model is described in Section 3.2.10.5 of Reference 3. With the submittal of Reference 2, the NRC has accepted the application for design certification (Reference 4).

RAI Number: SRXB.43

Question:

In SSAR Table 15.0-2: for event 15.1.3, pressure regulator failure-open, the maximum neutron flux approaches 243.9-percent nuclear boiling ratio (NBR). What is the change in minimum critical power ratio (MCPR) for this event? Doesn't this transient have an effect on MCPR? Why does a scram not occur at a flux level of 194-percent NBR (Hi: flux scram should occur at about 125-percent NBR).

GE Response:

The CPR change in the pressure regulator failed open event is 0.04. This value will be added to Table 15.0-2 in Amendment 1 (see attached).

Scram in this event occurs at 4.7 seconds when the water level swells to the L8 scram setpoint. The neutron flux has not increased at this time. It increases only later, after the turbine trips due to water level swell to level 9.

Table 15.0-2 Results Summary Of System Response Analysis Transient Events

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux (%NBR)	Max. Dome Pressure (MPa) Absolute	Max. Vessel Bottom Pressure (MPa) Absolute	Max. Steam Line Pressure (MPa) Absolute	Max. Core Average Simulate Heat Flux (%NBR)	Delta CPR	Frequency Category*	No. of Valves First Blow-down
15.1		Decrease in Core Coolant Temperature								
15.1.1		Loss of Feedwater Heating - 55.6°C (100°F)	110.2	7.171 (1040 psia)	7.263 (1053 psia)	6.985 (1013 psia)	110.2	0.12	a	0
15.1.2	15.1-2	Feedwater Controller Failure - Maximum Demand	194.0	7.694 (1116 psia)	7.813 (1133 psia)	7.667 (1112 psia)	109.6	0.11	a+	0
15.1.3	15.1-3	Pressure Regulator Failure - Open	243.9	7.137 (1035 psia)	7.262 (1053 psia)	6.969 (1011 psia)	100.0	** 0.04	a+	0
15.1.4		Inadvertent Opening of One SRV		SEE	TEXT					
15.1.6		Inadvertent RWCU/SDC Shutdown Cooling		SEE	TEXT					
15.2		Increase in Reactor Pressure								
15.2.1	15.2-1	Fast Closure of One Turbine Control Valve	117.7	7.772 (1127 psia)	7.895 (1145 psia)	7.631 (1107 psia)	113.0	0.16	a	0
15.2.1	15.2-2	Slow Closure of One Turbine Control Valve	117.7	7.773 (1127 psia)	7.896 (1145 psia)	7.633 (1107 psia)	112.7	0.16	a	0

RAI Number: SRXB.44

Question:

For event 15.2.1, pressure regulator failure closed, why is - critical power ratio (CPR), N/A in Table 15.0-2? There is a neutron flux increase to the Hi: flux scram setpoint which may indicate a reduction in CPR (Flux reaches 233-percent NBR).

GE Response:

The CPR change in the pressure regulator failed closed event is 0.14. This value will be added to Table 15.0-2 in Amendment 1 (see attached).

Table 15.0-2 Results Summary Of System Response Analysis Transient Events (Continued)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux (%NBR)	Max. Dome Pressure (MPa) Absolute	Max. Vessel Bottom Pressure (MPa) Absolute	Max. Steam Line Pressure (MPa) Absolute	Max. Core Average Simulate Heat Flux (%NBR)	Delta CPR	Frequency Category*	No. of Valves First Blow-down
15.2.1	15.2-3	Pressure Regulator Downscale Failure	233.4	8.168 (1185 psia)	8.282 (1201 psia)	8.171 (1185 psia)	110.4	~~N/A~~ 0.14	c	0
15.2.2	15.2-4	Generator Load Rejection, Bypass On	311.1	7.646 (1109 psia)	7.760 (1125 psia)	7.619 (1105 psia)	108.6	0.14	a	0
15.2.2	15.2-5	Generator Load Rejection with failure of all Bypass Valves	624.4	8.190 (1188 psia)	8.335 (1209 psia)	8.189 (1188 psia)	114.6	0.25	a+	0
15.2.3	15.2-6	Turbine Trip with Bypass On	314.1	7.647 (1109 psia)	7.761 (1125 psia)	7.620 (1105 psia)	108.7	0.14	a	0
15.2.3	15.2-7	Turbine Trip with failure of all Bypass Valves	624.4	8.190 (1188 psia)	8.335 (1209 psia)	8.189 (1188 psia)	114.6	0.25	a+	0
15.2.4	15.2-8	Inadvertent MSIV Closure	101.1	7.865 (1141 psia)	7.975 (1157 psia)	7.866 (1141 psia)	100.0	**	a	0
15.2.5	15.2-9	Loss of Condenser Vacuum	312.9	7.707 (1118 psia)	7.816 (1134 psia)	7.706 (1118 psia)	108.7	0.14	a	0
15.2.6	15.2-10	Loss of Non-Emergency AC Power	311.1	7.877 (1142 psia)	7.972 (1156 psia)	7.877 (1142 psia)	108.6	0.14	a	0
15.2.7	15.2-11	Loss of All Feedwater Flow	100.3	7.137 (1035 psia)	7.262 (1053 psia)	6.959 (1009 psia)	100.0	**	a	0

RAI Number: SRXB.45

Question:

For event 15.4.9, control rod drop accident, what would be the consequences if the separation-detection alarm failed for a stuck control rod, and it could drop to its maximum distance? Will the distance of the rod drop be limited so as to preclude unacceptable consequences?

GE Revised Response:

The system which detects control blade separation is designed and engineered as a safety-related system with redundant components. Therefore it is incredible that the blade could separate without detection. Independent from the control blade separation-detection alarm, the Rod Control and Information system (RC&IS) has a rod worth minimizer (RWM) design that restricts the maximum worth of an individual rod. The ganged withdrawal sequence of the RWM restricts the rod worth such that any unacceptable consequences are precluded. Best estimate calculations indicate that even if a rod drop occurred, the 280 cal/gm fuel enthalpy limit would not be exceeded.

RAI Number: SRXB.46

Question:

For event 15.5.1, inadvertent start-up of an isolation condenser, what is the single failure assumed for this event, other than the initiator? The SRP requires that an incident of moderate frequency in combination with any single active failure or operator error be considered.

GE Response:

No mitigation of this event or operator action is necessary, as described in the text, normal operation of the plant proceeds after the IC initiation. Active systems which are in operation prior to the initiating event are assumed to continue operating in the same mode. Applying an additional active failure or operator error would be equivalent to 2 simultaneous events initiating the transient.

RAI Number: SRXB.47

Question:

For the transients associated with a decrease in reactor coolant temperature, an increase in reactor pressure, and an increase in reactor coolant inventory, a single failure of a mitigative system should be assumed. These events should be analyzed in combination with any single component failure or single operator error.

GE Response:

The worst single active failure in this event would be failure of the Reactor Protection system to detect high APRM (Average Power Range Monitor) trip. This failure would not prevent scram since the RPS system is redundant and single failure proof.
Anticipated transients with additional failures are discussed and evaluated in Chapter 15. Usually, initiating events with additional multiple failures are analyzed to bound this type of event. These results are summarized in Chapter 15 of the SSAR and also in Table 15.0-2. Attached is a tabulation of these events.

For events not listed in the attached table, initiating events with additional failures are not quantitatively analyzed because either:

(1) No additional active single failure would cause tbe event to become more severe:

15.1.1 Loss of FW Heating

15.2.4 Inadvertent MSIV closure, or

(2) They are not limiting transients:

Other transients not listed in the attached table.

Initiating Event	Bounding Event With Additional Failures
15.1.2 Runout of one FW Pump	Runout of Two FW pump
Because of the triplicated, redundant digital controls, the single failure FW flow increase is bounded by this event.	(Section 15.1.2)
15.1.3 Opening of One Bypass Valve	Opening of All Control and Bypass Valves
Because of the triplicated, redundant digital controls, the single failure steam flow increase is bounded by this event.	(Section 15.1.3)
15.2.1 Closure of One Turbine Control Valve	Pressure Regulator Downscale Failure
Because of the triplicated, redundant digital controls, the single failure steam flow decrease is bounded by this event.	(Section 15.2.1)
15.2.2 Generator Load Rejection	Load Rejection with Failure of One Bypass Valve
(Section 15.2.2)	
	Load Rejection with Failure of All Bypass Valves
	(Section 15.2.2)
15.2.3 Turbine Trip	Turbine Trip with Failure of One Bypass Valve
(Section 15.2.3)	
	Turbine Trip with Failure of All Bypass Valves
	(Section 15.2.3)
Loss of One Auxiliary Power Transformer	Loss of All Non-Emergency Power
	(Section 15.2.6)