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MFN 06-178

Docket No. 52-010

June 16, 2006

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to NRC Request for Additional Information Letter No. 29 Related to ESBWR Design Certification Application – Control Rod Drive System Structural Materials/Reactor Internal Materials & Integrity of Reactor Coolant Pressure Boundary – RAI Numbers 4.5-1 through 4.5-32 and 5.2-6 through 5.2-29**

Enclosures 1 and 2 contain GE's response to the subject NRC RAIs transmitted via the Reference 1 letter. This completes GE's response to RAI Letter No. 29.

If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

D068

Reference:

1. MFN 06-156, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 29 Related to ESBWR Design Certification Application,* May 17, 2006

Enclosures:

1. MFN 06-178 – Response to NRC Request for Additional Information Letter No. 29 Related to ESBWR Design Certification Application – Control Rod Drive System Structural Materials/Reactor Internal Materials – RAI Numbers 4.5-1 through 4.5-32
2. MFN 06-178 – Response to NRC Request for Additional Information Letter No. 29 Related to ESBWR Design Certification Application – Integrity of Reactor Coolant pressure Boundary – RAI Numbers 5.2-6 through 5.2-29

cc:	WD Beckner	USNRC (w/o enclosures)
	AE Cubbage	USNRC (with enclosures)
	LA Dudes	USNRC (w/o enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRFs	0000-0054-4111 and 0000-0055-0405

Enclosure 1

MFN 06-178

Response to NRC Request for Additional Information Letter No. 29 Related to ESBWR Design Certification Application Control Rod Drive System Structural Materials/ Reactor Internal Materials RAI Numbers 4.5-1 through 4.5-32

NRC RAI 4.5-1

Design Control Document Tier 1 (DCD Tier 1), Figure 2.1.1-1, identifies chimney restraint, in-core housing, and drain line, which are parts of reactor internals, but which are not identified in DCD Tier 2, Table 4.5-1. Also, many reactor internal components identified on pages 2.1-3 to 2.1-5 of DCD Tier 1 are not mentioned in DCD Tier 2, Table 4.5-1. This discrepancy shows that DCD Tier 2 Table 4.5-1 is inadequate in identifying the core support structures and reactor internal components.

(A) Justify why Table 4.5-1 is adequate or revise Table 4.5-1 in DCD Tier 2 to include a list of all core support structures and reactor internal components with corresponding material selection.

(B) The components identified under the "Materials Used for the Core Support Structures" heading in Table 4.5-1 should not be considered as, or grouped with, the reactor internals because the core support structures have different ASME Code Class classification than some of the reactor internal components (e.g., non-ASME Code materials). Separating the core support structure components from the reactor internal components provides regulatory clarification. The staff recommends the title of DCD Table 4.5-1 be changed to "Material Specifications for Core Support Structures and Reactor Internals." Also, include a heading in DCD Table 4.5-1 (e.g., "Materials Used for the Reactor Internals") so that components listed under this heading would be identified as reactor internal components (e.g., steam dryers, steam separators, and chimney). The purpose of these two suggestions is to distinguish and separate the core support structure components from the reactor internal components.

GE Response

(A) Table 4.5-1 of Tier 2 will be revised to include all components mentioned on pages 2.1-3 through 2.1-5 of DCD Tier 1. The revised table includes material selection as shown in the following.

(B) Tier 2, Table 4.5-1 will also be revised to clearly distinguish between Core Support Structures as defined in the ASME Code, Section III, Subsection NG, Paragraph 1121 and Internal Structures defined in Paragraph 1122.

Table 4.5-1 will be replaced with the attached table in the next update of the DCD.

Table 4.5-1

Reactor Internals Material Specifications

Materials Used for the Core Support Structure:			
Component	**Form**	**Material**	**ASME Specification**
Shroud Support	Plate or Forging	Columbium[1] modified Ni-Cr-Fe Alloy	Modified Nickel Alloy 600 per ASME Code Case N-580-1
Shroud,	Plate	Stainless Steel[2]	SA-240, Type 304 / 304L / 316 / 316L
Core Plate	Plate and Forging	Stainless Steel[2]	SA-240 and SA-182/SA-182M, Type or Grade 304 / 304L / 316 / 316L
Top Guide	Plate and Forging	Stainless Steel[2]	SA-240 and SA-182/SA-182M, Type or Grade 304 / 304L / 316 / 316L
Peripheral Fuel Supports	Forging	Stainless Steel[2]	SA-182/SA-182M, Grade 304 / 304L / 316 / 316L
Orificed Fuel Support	Casting	Stainless Steel	SA-351/SA-351M, Grade CF3
Core Plate and Top Guide Studs, Nuts, and Sleeves	Bar	Stainless Steel[2]	SA-479/SA-479M, Type 304 / 304L / 316 / 316L and XM-19
Control Rod Drive Housing	Forging	Stainless Steel[2]	SA-336/SA-336M or SA-182/SA-182M Grade 304 or 316
Control Rod Guide Tube	Pipe, Bar and Forging	Stainless Steel[2]	SA-312/SA-312M and SA-479/SA-479M, Type 304 / 304L / 316 / 316L and XM-19
Control Rod Drive Penetration Stub Tubes	Forging	Columbium[1] modified Ni-Cr-Fe Alloy	Modified Nickel Alloy 600 per ASME Code Case N-580-1

[1] Also called Niobium in countries outside the U.S.
[2] Maximum carbon content limited to 0.02% except for castings

Table 4.5-1

Reactor Internals Material Specifications

Materials Used for the Internal Structures:			
Component	**Form**	**Material**	**ASME/ASTM Specification**
Chimney, Chimney Partitions and Chimney Restraints	Plate and Bar	Stainless Steel[2]	SA-240 and SA-479/SA-479M , Type 304 / 304L / 316 / 316L. The equivalent ASTM specification "A-" acceptable
Chimney Head and Steam Separator Assembly	Plate and Bar	Stainless Steel[2]	SA-240 and SA-479/SA-479M , Type 304 / 304L / 316 / 316L. The equivalent ASTM specification "A-" acceptable
Steam Separator	Pipe, Plate and Casting	Stainless Steel[2]	SA-312/SA-312M and SA-240, Grade or Type 304 / 304L / 316 / 316L; SA-351/SA-351M, Grade CF3. The equivalent ASTM specification "A-" acceptable
Chimney Head Bolts	Tube, Bar, Spring	Stainless Steel[2] and Ni-Cr-Fe Alloy	SA-213/SA-213M and SA-479/SA-479M, Type 304 / 304L / 316 / 316L; SA-479/SA-479M, Type XM-19; SB-637, N07750 (Alloy X-750). The equivalent ASTM specification "A- or B-" acceptable. Modified Nickel Alloy 600 per ASME Code Case N-580-1
Steam Dryer	Plate, Bar and Pipe	Stainless Steel[2]	SA-240, SA-479/SA-479M and SA-312/SA-312M, Type 304 / 304L / 316 / 316L; SB-637, N07750 (Alloy X-750). The equivalent ASTM specification "A- or B-" acceptable
Steam Dryer Seismic Blocks	Forging or Bar	Stainless Steel[2]	SA-182/SA-182M or SA-479/SA-479M, Grade XM-19. The equivalent ASTM specification "A-" acceptable
Feedwater Spargers	Pipe, Bar, Fitting and Casting	Stainless Steel[2]	SA-312/SA-312M and SA-479/479M and SA-403, Type, Grade or Class 304/304L/316/316L; SA351/351M, Grade CF3. The equivalent ASTM specification "A-" acceptable
Standby Liquid Control (SLC) Piping and Distribution Headers	Pipe, Bar or Plate	Stainless Steel[2]	SA-312/SA-312M and SA-479/SA-479M or SA-240, Type or Grade 304/304L/316/316L. The equivalent ASTM specification "A-" acceptable
In-Core Guide Tubes	Pipe	Stainless Steel[2]	SA-312/SA-312M, Grade 304 / 304L / 316 / 316L. The equivalent ASTM specification "A-" acceptable
In-core Guide Tube restraints	Plate, Strip and Bar	Stainless Steel[2] and Columbium[1] modified Ni-Cr-Fe Alloy	SA-240 and SA-479/SA-479M, Type 304/304L/316/316L. The equivalent ASTM specification "A-" acceptable. Modified Alloy 600 per ASME Code Case N-580-1
Guide Rod	Bar and Pipe	Stainless Steel[2] and Columbium[1] modified Ni-Cr-Fe Alloy	SA-479/SA-479M, SA-312/SA-312M, Type or Grade 304/304L/316/316L. The equivalent ASTM specification "A-" acceptable. Modified Nickel Alloy 600M per ASME Code Case N-580-1
Drain Line	Pipe, Bar or Plate	Stainless Steel[2]	SA-312/SA-312M and SA-479/SA-479M or SA-240, Type or Grade 304/304L/316/316L. The equivalent ASTM specification "A-" acceptable

[1] Also called Niobium in countries outside the U.S.

[2] Maximum carbon content limited to 0.02% except for castings

NRC RAI 4.5-2

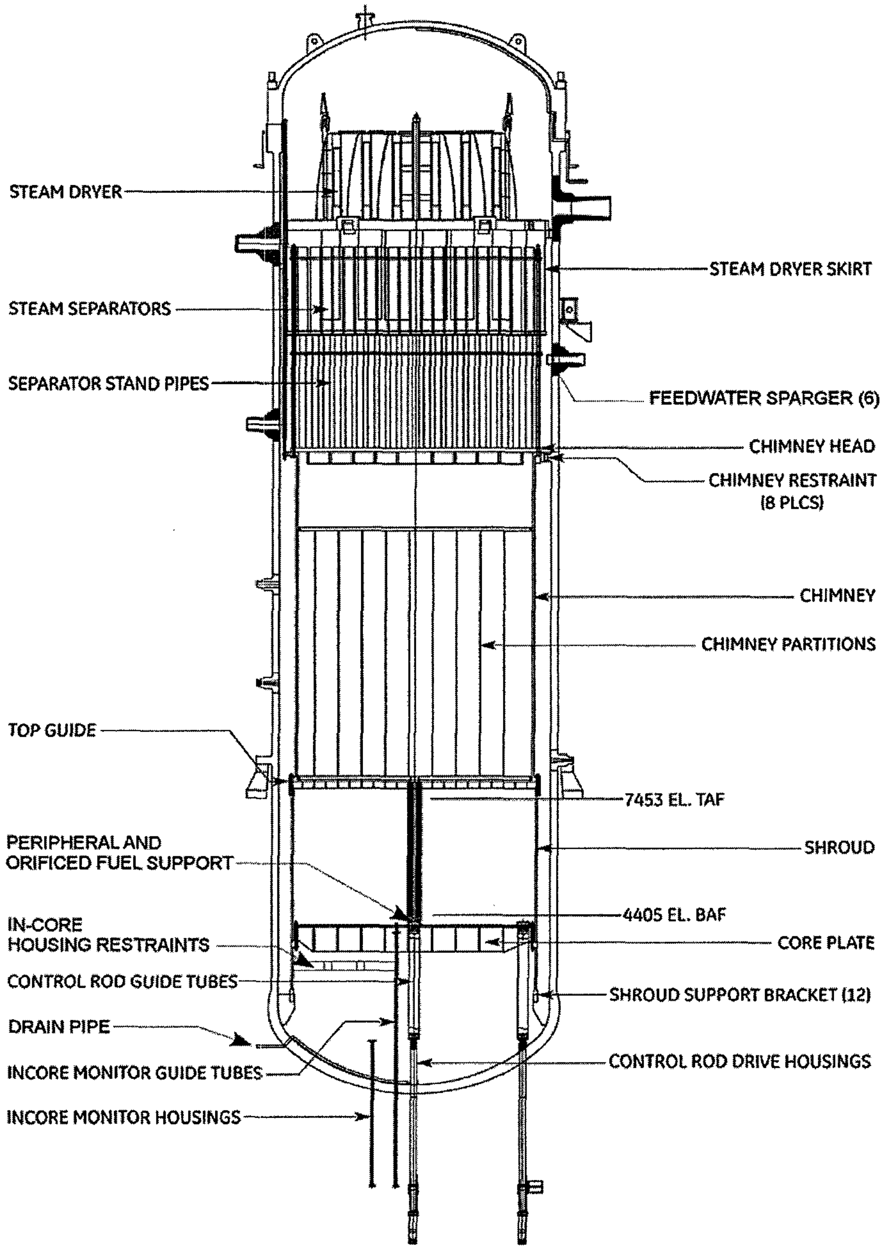
DCD Tier 2, Section 4.5.2 contains no drawings of the core support structures nor reactor internals. Figures 4.6-1 to 4.6-10 in DCD Tier 2 contain only schematics of control rod drive mechanisms. DCD Tier 1, Figure 2.1.1-1 contains a sketch of reactor internals without details. (A) Provide the detailed drawings and/or diagrams of all significant core support structures and reactor internal components. (B) Provide assembly drawings and diagrams to show how the core support structure components and reactor internal components are attached to each other and/or to the reactor vessel. (C) Please include the drawings and diagrams in Section 4.5.2 of DCD Tier 2.

GE Response

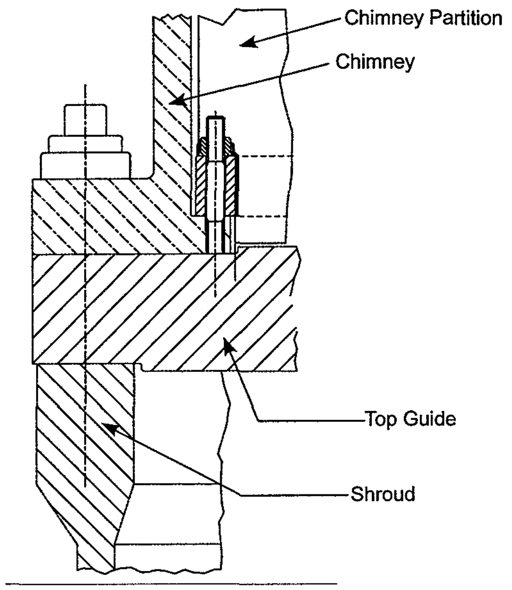
(A) To supplement Figure 2.1.1-1 a drawing of the reactor internals assembly is provided as shown in the following. This drawing shows the assembly of the major core structures and internal components listed in Table 4.5-1 contained in the response to RAI 4.5-1. Also, conceptual drawings of the shroud/top guide/chimney/chimney partition and the core plate-to-shroud joints are shown in the following. Sketches of other components are shown in responses to RAIs 4.5-18 Through 20.

(B) The attached assembly drawings will be included in Section 3.9.5 in the next update of the DCD.

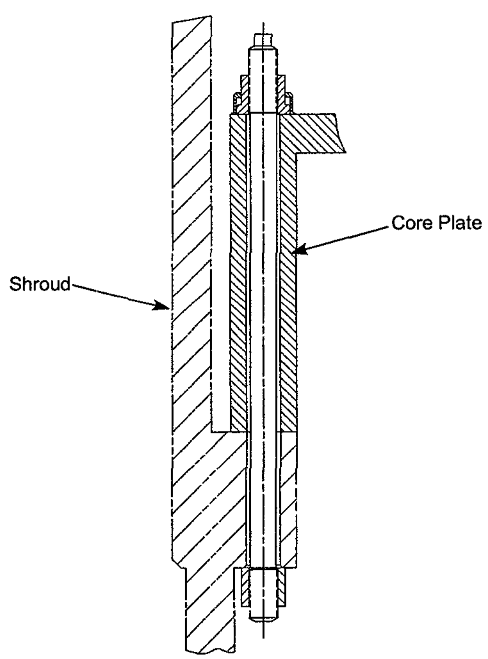
DCD Section 3.9 will be revised in the next update as described in (B) above.



ESBWR Reactor Assembly
Showing Reactor Internal Components



Typical Shroud, Chimney, Top Guide and Partition Assembly



Typical Core Plate to Shroud Connection

NRC RAI 4.5-3

DCD Table 4.5-1 identifies cast austenitic stainless steel, ASTM or ASME Grade CF3/CF3M, as a material used in the reactor internals and core support structures. Cast austenitic stainless steel is susceptible to the loss of fracture toughness due to thermal aging embrittlement, neutron irradiation embrittlement and void swelling in the reactor vessel. The staff's concern was documented in a letter from Christopher I. Grimes of NRC to Douglas J. Walters of Nuclear Energy Institute, subject: License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000 (ADAMS ML003717179). In addition, ultrasonic examinations of cast austenitic stainless steel have not been reliable. Address the loss of fracture toughness due to thermal aging embrittlement, neutron irradiation embrittlement and void swelling in cast austenitic stainless steel in the reactor vessel. Address the inspectibility of cast austenitic stainless steel by ultrasonic examinations.

GE Response

The use of cast stainless steel components in the ESBWR internals is very limited and confined to components that are common to previous BWR designs. As such, these components have more than 35 years of BWR operating experience with no known problems or failures. The one core support application is the fuel support casting. This is a removable, replaceable piece of hardware on which the fuel bundles sit. In this location the casting is well below the bottom of active fuel, and as such, sees relatively low neutron dose compared to other core support structures such as the shroud and top guide. Neutron induced void swelling does not occur because both the temperature and fluence are well below the nominal thresholds for this phenomenon in stainless steels[1]. Thermal aging is likewise not a concern. At the normal operating temperature for all BWRs of 550°F, thermal aging of low carbon stainless steel castings with less than 20% ferrite is barely measureable[2]. To assure the potential for thermal aging is thoroughly limited, GE specifies 20% maximum ferrite in the castings. The only other castings in the ESBWR internals design are the steam separator swirl generator casting and connector castings in the steam dryer. The swirler casting is a non-Code, non-safety related functional component that sees essentially no neutron dose because of its location. The only structural demands on this casting are to direct steam/water flow and support the minor weight of the individual separator assembly to which it is welded. As with the fuel support casting, this component is unchanged from early BWR designs except the carbon content is now limited to the L-grade range, and ferrite content is controlled to a range of 8-20%. The steam dryer castings are again non-Code, non-safety related components which see essentially no neutron dose. As with the castings mentioned above, they are low carbon with ferrite control. As such, they are highly resistant to thermal aging and stress corrosion cracking. Since none of these castings are subject to ultrasonic testing, either in fabrication or in-service, UT inspectability is not an issue.

No DCD changes will be made in response to this RAI.

References:

1. D. W. Sandusky, C. N. Spalaris, U. E. Wolff, "Irradiation Temperature Dependence of Void Formation in Type 304 Stainless Steel", Journal of Nuclear Materials, Vol. 42, (1972), pp 133-141.

2. O. K. Chopra and A. Sather, "Initial Assessment of the Mechanisms and Significance of Low-Temperature Embrittlement of Cast Stainless Steels in LWR Systems", NUREG/CR-5385, August 1990.

<u>NRC RAI 4.5-4</u>

DCD Table 4.5-1 indicates that niobium modified nickel-chromium-iron-alloy 600 per ASME Code Case No. –580-1 will be used in the shroud support. However, Code Case – 580-1 discusses adding columbium, not niobium, in Alloy 600. Discuss the discrepancy.

<u>GE Response</u>

Columbium and niobium are different names for the same element. Columbium is used mostly in the United States; in particular it has been adopted by the ASME Code. Niobium is the common name for this element in most other countries. A note describing this will be added to Table 4.5-1. See response to RAI 4.5-1.

DCD Section 4.5 will be changed as described in the next update.

NRC RAI 4.5-5

DCD Table 4.5-1 identifies that non-L grade 304 and 316 stainless steels are used for the reactor vessel internals and core support structures. Justify the use of non-L grade 304 and 316 stainless steels for the core support structures and reactor internals in light of the industry history of intergranular stress corrosion cracking in the 304 and 316 stainless steels and potential neutron irradiation embrittlement or irradiation assisted stress corrosion cracking in the BWR.

GE Response

The carbon content is limited not to exceed 0.02% in all welded wrought austenitic stainless steel components in the ESBWR that are exposed to reactor water at temperatures exceeding 93°C. The distinction in Table 4.5-1 between 304 and 304L/316 and 316L is only strength. As can be seen from the response to RAI 4.5-1, a note will be added to Table 4.5-1 stating that the carbon content of all type or grade 304/304L/316/316L used in the core support structures and reactor internal components is limited to maximum 0.02%.

A note saying that the maximum carbon content in stainless steel is limited to 0.02% except for castings will be added to Table 4.5-1 in the next update of the DCD.

NRC RAI 4.5-6

DCD Section 4.5.2.1 states that "...The other reactor internals are non-coded, and they may be fabricated from ASTM or ASME specification materials or other equivalent specifications...". The statement seems to be contradictory because "non-coded" generally means that a component is not fabricated using ASME specifications. (A) Clarify the above statement. (B) Identify the specific material specification for each of the reactor internal components and include this information in DCD Table 4.5-1.

GE Response

(A) Reactor internals that have a core support function are fabricated and certified to ASME Section III, Subsection NG-Core Support Structures. All other ESBWR internal components are considered "internal structures" consistent with Subsection NG terminology. For these components, materials may be procured to either ASTM or ASME Section II standards or equivalents. The individual ASTM and corresponding ASME material specifications are essentially identical, e.g. ASTM A 240 Type 316L plate is identical to ASME SA-240 Type 316L. As such, these are not considered "special materials". (B) Detail is provided in the response to RAI 4.5-1.

As described in the response to RAI 4.5-1, DCD Table 4.5-1 will be revised in the next update to show the requested details.

NRC RAI 4.5-7

(A) Discuss the operating experience (i.e., degradation) of the non-coded materials used in the reactor internals in the current BWR fleet. (B) Demonstrate that the non-coded material will provide the necessary strength, resistance to corrosion, and fracture toughness to maintain the safe operation of the ESBWR. (C) Discuss whether the non-coded components are designed for and analyzed with the same loading combinations per the ASME Code, Section III, as that used for the ASME Code components. If not, demonstrate by analysis that the failure of the non-coded components will not affect the structural integrity of the ASME Code components (e.g., the core support structures). (D) Clarify whether the non-coded components are considered as safety or non-safety category components.

GE Response

(A) As discussed in the response to RAI 4.5-6, the materials used for internal structures are identical in chemistry and properties to their Coded counterparts. Consequently, there is no distinction in behavior in BWR service between the Coded core support structures and other internal structures. (B) Strength, corrosion resistance, and toughness of the internal structure materials are equivalent to their Coded counterparts. (C) Internal structures are designed and analyzed using Article NG-3000 of ASME-III, Subsection NG as a guideline. Loading combinations are the same as those specified for core support structures. Stresses and fatigue usage factors will meet the limits specified in Subsection NG. (D) Internal structures may be safety-related or non-safety depending on their function. The standby liquid control (SLC) line is an example of a safety-related internal structure. Non-safety internal structures include such components as the steam separators and steam dryer.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-8

Discuss the industry standards that the non-coded components will follow in terms of material selection, fabrications (including welding), construction, design (e.g., stress analysis), testing, and inspections.

GE Response

Material selection, fabrication, etc. are consistent with ASME Code. Welding procedures and welders are qualified to ASME Section IX. Inspection methods are consistent with ASME Section V and acceptance criteria follow Subsection NG. See also response to RAI 4.5-7(C) with respect to design and analysis.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-9

(A) The title of DCD Section 4.5.2.2 is "Controls on Welding". However, welding is not explicitly mentioned in this section. To clarify the intent, revise the first sentence in Section 4.5.2.2 to read: "...Core support structures are welded in accordance with...". (B) Standard Review Plan (SRP) Section 4.5.2.II.2, Draft Revision 3, April 1996, specifies that methods and controls for core support structures and reactor internals welds must be in accordance with the ASME Code, Section III, Division 1, NG-4000, and the welds must be examined and meet acceptance criteria as specified in NG-5000. The second sentence in DCD Section 4.5.2.2 discusses the welding of the reactor internals without referring to the above ASME Code sections. Justify why the welding of the reactor internals does not follow ASME Code, Section III, NG-4000 and NG-5000, and ASME Code, Section IX. (C) Identify the core support structure and reactor internal components that require welding and describe the welding technique/procedures that will be used.

GE Response

(A) Note that it is intended that "fabrication" as used in this paragraph encompasses all fabrication processes, including welding as defined in ASME Section III, Article NCA-9000. (B) For core support structures the components are required to be built and certified in full compliance with ASME Section III, Subsection NG. Therefore compliance with NG-4000 and NG-5000 is implicit and all welding will be performed and inspected accordingly. It was not considered necessary in the DCD section to explicitly refer to these portions of NG since full compliance with Subsection NG in its entirety is required. For the non-code internals, welding qualification according to ASME Section IX is required. Welding practices and inspections are generally consistent with NG-4000 and NG-5000. (C) Most of the core support structures and reactor internals require some welding for assembly. The main exceptions are the fuel supports that rest on the core plate, which are machined from forgings or castings. Welding processes will be those commonly applied to stainless steels and nickel alloys such as SMAW, GTAW, SAW, GMAW, etc. Both manual and automatic processes will be applied. The specific welding techniques and procedures cannot be defined at this time because those details are dependent on the facility contracted to do the fabrication work.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-10

SRP Section 4.5.2.II.3, Draft Revision 3, April 1996, specifies that "...the acceptance criteria of the nondestructive examination shall be in accordance with the requirements of ASME Code, Section III, Division I, NG-5300..." DCD Section 4.5.2.3 does not specify the acceptance criteria for the nondestructive examination. Revise DCD Section 4.5.2.3 to include the acceptance criteria for nondestructive examination and identify the appropriate ASME Code section or justify why the acceptance criteria are not needed.

GE Response

As with the discussion of welding above, for core support structures, full compliance with Section III, Subsection NG is a given and so stated. Likewise, for the internal components that have a pressure retaining function, full compliance with NB is required and so stated. Therefore it was not considered essential in the DCD to explicitly reference individual paragraphs such as NB/NG-5300.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-11

DCD Section 4.5.2.3 discusses the nondestructive examinations of control rod drive housings and peripheral fuel supports. SRP Section 4.5.2.I.3, Draft Revision 3, April 1996, recommends that each product form in the reactor internals and core support structures be examined. (A) Justify why product forms other than control rod drive housings and peripheral fuel supports do not need nondestructive examinations or revise the title and contents of DCD Section 4.5.2.3 to require nondestructive examinations of all product forms in the reactor internals and core support structures. (B) Identify the specific tubular products that will be hydrostatically tested.

GE Response

Section 4.5.2.3 will be revised to reflect the expanded scope of Draft Revision 3 to SRP 4.5.2.I.3. Examination of core support structure materials and welds will be in full compliance with ASME Section III, Subsection NG. Pressure retaining components and welds will likewise be inspected in full compliance with Subsection NB.

DCD Section 4.5 will be revised in the next update as described in the following:

4.5.2.3 Non-Destructive Examination

Materials for core support structures will fully conform and be certified to ASME Section III, Subsection NG. Examination of materials (examination methods and acceptance criteria) are specified in NG-2500. Examination methods and acceptance criteria for core support structure weld edge preparations and welds are provided in NG-5000. Tubular products that are pressure boundary components (CRD and in-core housings) will be examined according to ASME Section III, NB-2500, and associated pressure retaining welds will be examined according to NB-5000. For non-ASME code reactor internal structures and associated welds, examinations are established based on relevant design and analysis information, and take guidance from NG-2500 and NG-5000 respectively.

<u>NRC RAI 4.5-12</u>

DCD Section 4.5.2.4 states that significantly cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers. There are advantages and disadvantages of using cold-worked materials in certain applications. Justify the use of cold-worked materials in vanes considering the adverse impact of the cold work on the microstructure of the material and its susceptibility to stress corrosion cracking.

<u>GE Response</u>

Some degree of cold working is necessary to form the steam dryer vane shape. This design is essentially unchanged from the earliest BWRs. As such, there is over 35 years of operating experience with this design and no failures of vanes have been observed. The material has been updated to current low carbon standards, and maximum hardness is controlled to a level well below the threshold for SCC. Since the only function of the vanes is to direct steam flow, there is virtually no sustained tensile stress on these parts. Also note that even if SCC were to occur, there is virtually no potential to create a loose part because the vane banks are contained between perforated plate assemblies.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-13

DCD Section 4.5.2.5 states that "...Alloy X-750 components are fabricated in the annealed and aged condition. Where maximum resistance to stress corrosion is required, the material is used in the high temperature (1093°C) annealed plus single aged condition...". Alloy X-750 materials are susceptible to intergranular stress corrosion cracking due to equalized and aged heat treatment conditions (Reference: BWRVIP-41, "BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," EPRI 1012137, page 2-3). (A) Identify the reactor internal components fabricated from Alloy X-750. (B) Provide information on aging heat treatment process of Alloy X-750 (i.e., aging temperature and holding time). (C) Justify how this aging process will help to prevent/minimize stress corrosion cracking. (D) Provide the optimal hardness value that is required to minimize the susceptibility to stress corrosion cracking. (E) Discuss the discrepancy in Alloy X-750 not being identified in DCD Table 4.5-1 although this material is identified in DCD Section 4.5.2.5 as a material used in reactor internals.

GE Response

(A) Other than the CRD components identified in DCD Section 4.5.1, use of Alloy X-750 in ESBWR internal components is very limited. The only application positively identified at this time is a coil spring on the shroud head bolt, a non-safety related component, and possibly a latch component in the steam dryer. (B) Age hardening will be a single step treatment. Age hardening of coil spring noted above will be for 16 hours at 732°C (1350°F). Any other shapes of X-750 will be age hardened at 704°C for 20-21 hours. (C) Actually it is the annealing process prior to the aging treatment that bestows most of the improved SCC resistance to X-750. ESBWR X-750 components will be annealed at 1080 - 1108°C (1975 - 2025°F). This is the HTA (high temperature anneal) condition developed for improved BWR jet pump beam performance in the early 1980s. It is consistent with the Type 3 heat treatment of ASTM/ASME B/SB-637 and the EPRI Guidelines on X-750. The HTA treatment in conjunction with a single step aging treatment is considered to provide optimum stress corrosion resistance to X-750 in BWR applications. (D) Hardness has not been identified as a control parameter for SCC resistance except that it is known that hardness in excess of Rockwell C40 can indicate elevated susceptibility. B/SB-637 Type 3 heat treatment specifies a RC40 maximum hardness. (E) Per the response to RAI 4.5-1, the table will be modified to list the known uses of X-750.

DCD Section 4.5 will be revised in the next update as described in (E) above.

NRC RAI 4.5-14

(A) Discuss the pre-service inspection and inservice inspection program of all core support structure and reactor internal components. For each component, the discussion should include specific examination technique, frequency of the inspection, acceptance criteria, the area/coverage of the inspection, and the industry codes/requirements used. (B) Provide a list of components that will not be inspected during the pre-service inspection or inservice inspection activities and explain why the inspection is not needed.

GE Response

(A) The pre-service and inservice inspections of core support structures and internal components are an owner/COL issue. Visual examination of the core support structures will be performed during plant outages as required by the ASME Boiler and Pressure Vessel Code, Section XI, Table IWB-2500-1, Item B13.40. The frequency of the examinations will be at least as outlined in Subarticle IWB-2400. The examination personnel shall be qualified per Subarticle IWA-2300.

(B) There are no ASME Code requirements for pre-service and inservice inspections of reactor internal components. These components include:

- Chimney, Chimney Partitions and Chimney Restraints
- Chimney Head and Steam Separator Assembly
- Chimney Head Bolts
- Steam Dryer Assembly
- Feedwater Spargers
- Standby Liquid Control (SLC) Piping and Distribution Headers
- In-core Guide Tubes
- In-core Guide Tube restraints
- Guide Rods
- Drain Pipes

During the fabrication of core support structures all material is examined as required by Subarticle NG-2500 of the ASME Code, Section III, Subsection NG. For the examination of internal components the ASME Code is used as a guideline. A liquid penetrant examination is required on the weld prep surfaces prior to welding and on all machined surfaces. The extent of non-destructive examination of welds is determined by the weld quality and fatigue factors (ASME III, Table NG-3352-1) applied to the weld joints in the design analysis.

All welds, materials and subassemblies not accessible for inspection in the completed assembly are inspected for quality and cleanliness prior to the last operation that results in inaccessibility.

A visual examination of the completed components that meets the requirements of the ASME Code Section XI, Subsubparticle IWA-2210 is performed in the shop to serve as "pre-service visual inspection".

The same rigorous quality and cleanliness requirements are applied to the installation of the reactor internals in the field.

No DCD change will be made in response to this RAI.

NRC RAI 4.5-15

The BWR Vessel and Internals Project (BWRVIP) has published many guidelines in BWRVIP reports related to the reactor internals. The NRC has approved some of the BWRVIP reports. Discuss briefly which guidance/reports will be used for which components in the ESBWR. If none of the BWRVIP guidelines will be followed as a matter of practice or policy, provide explanation.

GE Response

The BWRVIP Guidelines were written for maintenance, inspection, and repair of currently operating BWRs and do not address new plant construction. Consequently, these guidelines are not specifically used to establish ESBWR requirements. However, ESBWR materials selection and controls are generally consistent with the ALWR Utilities Requirements Document (URD), also published by EPRI.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-16

Discuss the maintenance program for the bolts and threaded fasteners used in the core support structures and reactor internals to ensure their structural integrity (i.e., prevent bolt cracking) and to prevent them from becoming loose parts in the reactor coolant system.

GE Response

Cracking of bolts and fasteners in core structure components and reactor internal components has not been an issue in operating BWR plants. Positive locking mechanisms are used for bolting applications; e.g., nuts are tack welded in place to prevent them from coming loose and visual inspection is performed during installation. Austenitic stainless steel bolts and nuts of type 304/304L/316/316L were generally used in the past, but in newer plants, including ESBWR, nitrogen strengthened austenitic stainless steel Grade XM-19, material is being used for highly loaded bolted joints.

Since there is no ASME Code, Section XI requirement for inservice inspection, and based on favorable BWR operating experience, there is no formal ESBWR maintenance and inspection requirements for bolts and threaded fasteners inside the reactor pressure vessel.

No DCD change will be made in response to this RAI.

NRC RAI 4.5-17

(A) Identify the ASME Code requirements for material selection, inspection, design, fabrication, and construction of the chimney, chimney partitions, and chimney head. (B) Describe the fabrication, assembly, and installation of the chimney, chimney partitions, and chimney head. (C) Discuss whether a mockup test of the chimney assembly in a reactor vessel environment has been conducted to verify the structural integrity of the chimney assembly.

GE Response

(A) The ASME Code, Section III, Subsection NG is used as a guideline for the material, design, fabrication and inspection of the chimney, chimney partitions and chimney head. These components are classified as internal structures and do not require an ASME NPT Code Stamp.

(B) The chimney partition assembly consists of a grid of square structures, each of which encompasses 16 fuel assemblies, and a bottom and a top ring. The bottom ring rests on and is pinned and bolted to the bottom flange of the cylindrical chimney shell. The top ring of the assembly is supported against the inside of the chimney shell. The chimney assembly is bolted to the top guide and laterally supported by eight chimney restraints at the top.

(C) With reference to DCD Tier 2, Appendix 3L, an air and water two-phase flow vibration test of both a 1/6-scale and a 1/12-scale model of a single chimney cell was performed. The results of the scale testing were extrapolated by a two-phase flow analysis to determine the characteristics of the pressure fluctuations acting on the partition wall of a full size cell in steam-water conditions. Stress analysis based on the test results showed an adequate margin against the allowable vibration peak stress amplitude.

No DCD change will be made in response to this RAI.

NRC RAI 4.5-18

The core shroud supports in the current BWR fleet are supported from and attached to the bottom of the reactor. The ESBWR design is attached and supported at the side wall of the reactor vessel which may result in a bending moment on the vessel wall. (A) Discuss whether the design of the core shroud supports considered the potential bending of the reactor vessel wall (i.e., the shroud supports may not sustain the loads as calculated in the structural analysis because the vessel wall may not be as rigid as assumed in the analysis). (B) Discuss whether the stress analysis of the reactor vessel shell considered the bending moment generated by the core shroud supports. (C) The core shroud supports use Niobium-modified Inconel 600 alloy. Alloy 600 material is susceptible to stress corrosion cracking. Justify the selection of this material in the reactor vessel or describe the design features that will be used to mitigate stress corrosion cracking. (D) Provide the drawings and design details including the location and installation of the core shroud supports.

GE Response

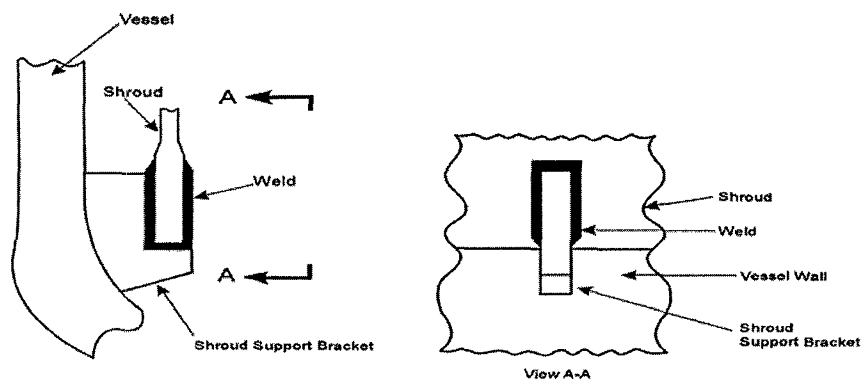
(A) Shroud supports attached directly to the reactor vessel wall were used in the past in vessels built by Combustion Engineering; e.g., Plant Hatch. Analyses and experience have proven that the bending stresses produced by the cantilever shroud support design in these vessels are acceptable.

(B) The bending moment from the shroud support will be included in the ESBWR reactor vessel stress analysis. Since the moment from the ESBWR shroud support is smaller than that in the aforementioned vessels due to a smaller gap between the shroud and the vessel wall, excessive bending stresses are not expected.

(C) The core support material is Ni-Cr-Fe Alloy 600 with columbium added. (Note columbium is frequently called niobium outside the U.S.). Use of this material is permitted by ASME Code Case N-580-1. Columbium modified Ni-Cr-Fe Alloy 600 has been successfully used in the ABWRs, and tests have shown it is highly resistant to stress corrosion cracking in a BWR environment.

(D) A sketch showing the connection between the shroud and the shroud support is shown below.

No DCD change will be made in response to this RAI.



Typical ESBWR Shroud-To-Shroud Support Joint.
12 Support Brackets Spaced Equally Around the Reactor Vessel Wall

NRC RAI 4.5-19

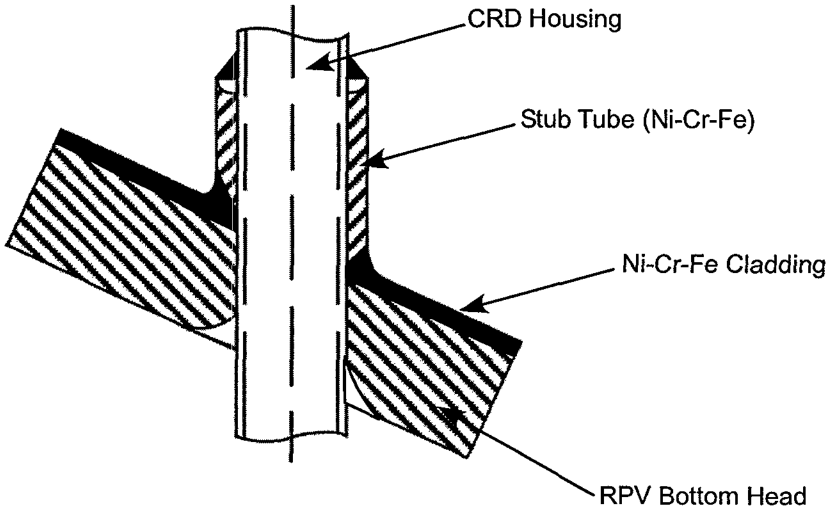
DCD Tier 1, Page 2.1-2, states that "...The control rod drive (CRD) housings... are welded to CRD penetrations (stub tubes) formed in the reactor pressure vessel (RPV) bottom head...". DCD Tier 1, Page 2.1-4, states that "...the upper end of the CRD housing is welded to a stub tube that is directly welded to the bottom of the vessel...". It is not clear whether the stub tube is welded or formed as part of RPV bottom head forging. (A) Provide assembly drawings of the CRD housing and stub tube to show how they are attached to each other and to the bottom of the vessel. (B) Discuss weld joint details, welding processes, post-weld heat treatments, materials to be used, and the fabrication sequence that will be used to prevent sensitization of the stainless steel material (based on operating experience in the current BWR fleet, i.e., Oyster Creek).

GE Response

(A) A schematic drawing of the reactor vessel CRD penetrations is shown below. As can be seen from the drawing, the stub tubes are welded to the Ni-Cr-Fe cladding in the bottom head. The stub tube material is columbium modified Ni-Cr-Fe Alloy 600 per ASME Code Case N-580-1.

(B) Welding of the joints between the stub tubes and the bottom head and between the CRD housings and the stub tubes is performed with a process using Nickel Alloy 82 filler material according to ASME SFA-5.14, Grade ER NiCr-3 (use of Alloy 182 according to SFA-5.11 Grade E NiCrFe-3 is not permitted). The final post weld heat treatment of the vessel is performed after the Ni-Cr-Fe stub tubes are welded into the bottom head. This type of stub tube connection and material has successfully been used in the recent ABWRs. Oyster Creek type stainless steel stub tubes are not used any more in the BWRs.

No DCD change will be made in response to this RAI.



Control Rod Drive (CRD) Penetration

NRC RAI 4.5-20

Section 2.1.1 in DCD Tier 1 states that a latticework of clamps, tie bars, and spacers provide lateral support and rigidity to the in-core guide tubes. (A) Provide assembly drawings of the lateral support components, in-core guide tubes, how the lateral support components are inter-connected, and how the in-core guide tubes are attached to the shroud. The drawings should be included in Section 4.5.2 in DCD Tier 2. (B) Identify materials used for the lateral support components and in-core guide tubes. (C) Identify the number of penetrations.

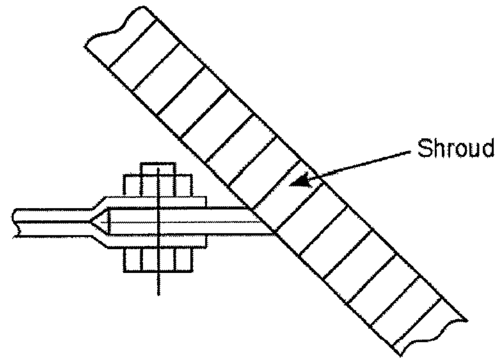
GE Response

(A) The lower ends of the in-core guide tubes are welded to the in-core housings in the bottom of the reactor vessel. The top ends extend through holes in the core plate that provides lateral support. Conceptual drawings illustrating the interconnections between the in-core guide tubes lateral supports and their attachments to the lower portion of the shroud, and the connections between the guide tubes and the core plate will be included in Section 3.9.5 of DCD Tier 2. The figures to be included are shown below.

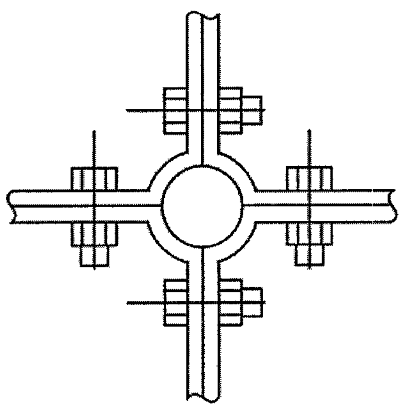
(B) The material of the lateral supports and the in-core guide tubes is shown in the revised Table 4.5-1 contained in the response to RAI 4.5-1.

(C) There are a total of 88 in-core penetrations in the reactor vessel bottom head. The locations of the penetrations within the core are shown on Figures 7.2-6 and 7.2-7 of DCD Tier 2.

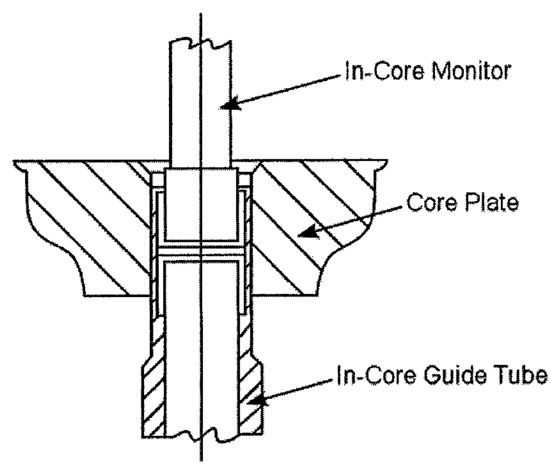
DCD Section 3.9 will be revised in the next update as described in (A) above.



Typical Lateral Support Connection To Lower Shroud



Typical Inter-Connection Between
Incore Guide Tube Lateral Supports



Typical Connection Between In-Core
Guide Tube and Core Plate

NRC RAI 4.5-21

Discuss whether a hydrogen water chemistry program will be implemented in the reactor vessel to mitigate stress corrosion cracking. If so, discuss briefly the hydrogen water chemistry program and associated requirements.

GE Response

It should be noted that materials selection and process controls are defined without taking any credit for application of hydrogen water chemistry. That is, it is intended the internal components are capable of operating for the design life of ESBWR without experiencing stress corrosion cracking failures. Nevertheless, there are some unknowns and unforeseen circumstances in contemplating 60 years of operation. Consequently, hydrogen water chemistry may be adopted primarily for added margin and will be done at the Owner's option. The ESBWR design does incorporate features (injection taps, etc.) that facilitate installation of the HWC system either before or after initial startup.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-22

DCD Tier 1, Page 2.1-5, states that special controls on material fabrication processes are exercised when austenitic stainless steel is used for construction of RPV internals. Describe the special controls that are used for material fabrication.

GE Response

The controls referenced above are contained in the detailed purchase specifications used to procure materials and fabrication of components. Consequently, the full level of detail is not yet in place for ESBWR. However, when these documents are prepared, the content will be very similar to existing specifications for ABWRs. For preparation of individual equipment documents guidance will be taken from "MATERIALS AND PROCESSES CONTROLS", a top level ESBWR materials and process document. The general practice is to have a materials specification that is used in conjunction with a fabrication specification for individual groups of equipment. For stainless steel materials there are a number of controls that are placed on the supplier that are more detailed than the basic ASTM/ASME requirements. In addition to the 0.02% maximum carbon limitation that will be included in the revised DCD Table 4.5-1, these equipment requirements documents will include the following among the controls generally applied:

- Limitations on cobalt content (varies depending on proximity to the core).
- Detailed controls on heat treatment time/temperature and quenching.
- For Nuclear Grade 304/316, confirmatory test of yield strength at 288°C (550°F).
- Control of maximum hardness.
- Sensitization Test (Modified ASTM A 262 Practice A).
- Intergranular Attack control.
- Limitations/Controls on weld repairs
- Cleaning, marking, and packaging controls.

Fabrication of stainless steel components is likewise controlled with a detailed fabrication specification that includes such features as:

- Control of hardness, including:
    - Control of mechanical cutting methods
    - Machining controls
    - Grinding controls
    - Controls on cold bending, forming and straightening
    - Limitations on final hardness (both bulk and surface)
- Control of thermal processes such as:
    - Thermal cutting methods and heat input
    - Hot forming and bending
    - Specific controls of induction bending
- Welding Controls including:
    - Joint configurations, fitup and gap, alignment

  - Permitted processes
  - Heat input control
  - Backpurge and flux controls
  - Allowed filler metals
  - Ferrite control and measurement method
  - Weld metal control and storage
  - Restricted access qualification (R.G. 1.71)
- Control of repairs including allowed weld repairs
- Detailed NDE requirements
- Cleaning and Cleanliness controls including control of miscellaneous process materials
- Traceability of material, marking, and packaging for shipment.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-23

DCD Tier 1, Page 2.1-4, discusses the feedwater spargers. Cracking of the feedwater spargers in the current BWR fleet is discussed in the NRC report, NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." Describe design features, fabrication processes, and water chemistry to minimize or prevent cracking in feedwater nozzles and spargers in ESBWR. Discuss the inspection program for the feedwater spargers (examination scope, technique, and frequency).

GE Response

Cracking of the feedwater spargers in some of the earlier BWRs was caused by three mechanisms; namely, high cycle thermal fatigue due to sub-cooled water leaking through the loose fit between the feedwater nozzle and the thermal sleeve, sub-cooled water shedding from the sub-cooled thermal sleeve periodically cooling the nozzle, and thermal stratification in the feedwater sparger during low flow.

In the ESBWR the feedwater sparger, thermal sleeve and vessel nozzle are welded together, thus eliminating the leakage flow of sub-cooled water. To prevent the vessel nozzle from being exposed to cold water shedding from the thermal sleeve carrying sub-cooled water, a double thermal sleeve of a tuning fork design is used. The sub-cooled feedwater flows through the inner sleeve that is welded to the sparger. The concentric outer sleeve protects the vessel nozzle from being exposed to the cold water periodically shedding from the outer surface of the inner sleeve. The tuning fork design mitigates the thermal stresses between the austenitic stainless steel thermal sleeve and the low alloy vessel nozzle. The ESBWR feedwater sparger has a row of spray nozzles mounted at the top of the sparger pipes so that the sparger will always be filled with water from the feedwater piping system with minimal mixing with the warmer reactor water vessel water. This sparger design helps to minimize thermal stratification within the sparger and piping during low flow conditions.

This sparger/thermal sleeve design has successfully been used in the recent BWR product lines as well as in retrofit designs installed in Monticello and Tsuruga-1 in the early 1980s. In regard to in-service inspections it is referred to in the response to RAI 4.5-14.

No DCD change will be made in response to this RAI.

NRC RAI 4.5-24

DCD Tier 1, Section 2.9, and DCD Tier 2, Section 4.4.5, describe the loose parts monitoring system to detect metallic parts in the reactor vessel. The system uses acoustic sensors to detect loose parts and alarms to notify the operators. There is no discussion on the measures that will be taken to prevent the generation of loose parts. Describe the programs that will be used to prevent and manage metallic loose parts in the reactor vessel during fabrication/assembly, maintenance, normal operation, and refueling activities.

GE Response

Fabrication and installation of the reactor vessel and the reactor internals is performed per a quality program that meets the requirements of the Code of Federal Regulation, 10CFR50, Appendix B. This includes implementation of a cleanliness program.

Prior to plant operation flushing of the vessel and attached piping will be performed to remove debris that have may have collected during construction. Loose part control during service and maintenance will be implemented by the plant owner.

No DCD change will be made in response to this RAI.

NRC RAI 4.5-25

Discuss the likelihood of the following degradation mechanisms affecting all significant core support structures and reactor internal components: ductile and brittle fractures, fatigue failures, distortion failures, wear failures, erosion (cavitation and liquid-impingement), corrosion (pitting, leaching, galvanic, and intergranular), creep, corrosion-fatigue, hydrogen-damage failures, embrittlement (neutron irradiation and thermal), and stress corrosion cracking. Discuss the steps taken to minimize or prevent the applicable degradation mechanisms (e.g., inspection and mitigation techniques).

GE Response

Prior to considering individual degradation mechanisms, it should be noted that ESBWR is an evolutionary design that incorporates many aspects of prior BWR designs. In particular the operating environment to which internal components are exposed is essentially identical. Therefore, over 30 years of operating experience can be used to determine which degradation mechanisms may be active in ESBWR. On that basis, the individual mechanisms noted in RAI 4.5-25 are discussed below:

- Ductile and brittle fractures- Use of ASME Code design rules assure that there is no risk of ductile failures even under upset conditions. Stainless steels and nickel alloys are not embrittled by fabrication processes, thermal aging, or exposure to BWR water. Although neutron irradiation decreases ductility, at the highest exposure levels for reactor internals, significant residual toughness is retained.
- Fatigue failures- For the most part, fatigue failures have been very limited in BWR internals with a few exceptions. Historically some fatigue failures have occurred in jet pump components, but these do not exist in the ESBWR design. The other component that has experienced fatigue issues in operating BWRs is the steam dryer. Potential for fatigue failures in the ESBWR steam dryer is being addressed by implementation of a highly fatigue resistant design based on extensive finite element and CFD modeling, along with scale model testing.
- Distortion failures- No distortion failures have been observed in operating BWR internals with one exception. A series of steam dryers were fabricated with thin (0.125 inch) end hood plates, which became distorted by a pressure pulse generated by rapid MSIV closure. This was corrected by replacement with thicker hood material. The ESBWR steam dryer end hood plates are thicker yet.
- Wear failures- Other than the control rod drives and control blades, there are no moving parts in the ESBWR reactor internals. Wear has been considered by choosing hardfacing or wear resistant alloys for moving components subject to wear. All the moving components that would potentially be subject to wear are routinely removable and replaceable.
- Erosion (cavitation and liquid-impingement)- These degradation phenomena have not been observed in the internals of operating BWRs and are not expected in ESBWR. Stainless steels, because of the high chromium content, are very resistant to erosion.

- Corrosion (pitting, leaching, galvanic, and intergranular)- Stainless steels and nickel alloys have not been observed to experience these corrosion phenomena in the BWR environment (very pure deionized water).
- Creep- Stainless steels and nickel alloys do not experience creep at the maximum operating temperature of ESBWR.
- Corrosion-fatigue- A corrosion-fatigue interaction has not been observed in BWR internal components. The fatigue failures noted above are considered to have been the result of cyclic loading without any apparent or significant environmental factor. In any case, this concern has been addressed by design improvements in ESBWR to eliminate potential for fatigue failures.
- Hydrogen-damage failures- Hydrogen driven failure mechanisms such as hydriding are not active in the BWR environment. Stainless steels and nickel alloys are not susceptible to hydrogen embrittlement or hydriding under the thermodynamic conditions in BWR water, even for a plant operating on Hydrogen Water Chemistry.
- Embrittlement (neutron irradiation and thermal)- Stainless steels and nickel alloys are not subject to thermal embrittlement at the ESBWR operating temperature (288°C). Stainless steel does experience a loss of ductility and toughness with neutron irradiation. This loss becomes significant at irradiation doses exceeding about $1x10^{21}$ n/cm^2 (E > 1 Mev). However, only certain areas of the reactor internals receive neutron dose exceeding this level and even at the maximum dose for internals some significant degree of toughness is maintained. A special limitation on design loads is placed on reactor internal components that are subject to high radiation levels. Operating BWRs achieve similar dose levels in reactor internals and no embrittlement failures have been observed, even in plants where there is frequent seismic activity.
- Stress corrosion cracking- Potential for SCC of reactor internals has been addressed for ESBWR by 1) using only solution annealed, low carbon stainless steels and nickel alloys modified for high SCC resistance, 2) strict control of fabrication and installation processes (For details see Response to RAI 4.5-22.), and 3) application of polishing to remove surface cold work in the weld heat affected zones of the major structural welds in the large internals. These measures are expected to greatly reduce the potential for SCC of internals in ESBWR relative to the currently operating BWRs. Routine in-service inspections will monitor the condition of the internals and be capable of detecting any degradation by SCC in the unlikely event that it occurs.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-26

DCD Section 4.5.1.1 "Materials" indicates that the applicant used Regulatory Guide (RG) 1.85 for the design of the ESBWR. However, RG 1.85 was withdrawn by the NRC on June 10, 2004. RG 1.84 contains guidance on all Section III Code Cases, including those related to materials and related testing in Division 1 which were previously contained in RG 1.85. Please update Section 4.5.1.1 to reflect the correct Regulatory Guide used in the design of the ESBWR.

GE Response

DCD Section 4.5 will be revised in the next update to reference RG 1.84 in Subsection 4.5.1.1 instead of RG 1.85.

NRC RAI 4.5-27

Please discuss the selection, basis for selection, and operating experience with the materials selected and used in the Cobalt bearing and non-Cobalt bearing hard surfacing alloys in the ESBWR design.

GE Response

Other than the cobalt bearing materials in the FMCRD noted in DCD Section 4.5.1, no cobalt bearing alloys are used in the ESBWR internals design. The components in the FMCRD are small bearings and other parts where maximum wear resistance is required. Because these materials are contained within the CRD, they are not directly activated because of being located far below the bottom of active fuel where neutron fluence is minimal. Release of cobalt to reactor water by general corrosion is very limited because the operating temperature inside the drive is substantially lower than reactor temperature, flow rates are low, and these cobalt base alloys have generally high corrosion resistance. The non-cobalt alloys used in wear and hard surfacing applications in the FMCRD components were selected specifically to minimize the use of cobalt base alloys. These alloys were qualified for the FMCRD application by extensive mockup testing for ABWR and have been in service in Kashiwazaki-Kariwa 7 since it started up in 1997. Any of these components are readily replaceable as part of routine CRD maintenance.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-28

Given the relationship between mechanical and thermal processing of Alloy X-750 and its susceptibility to stress corrosion cracking, please provide a basis for the heat treatment described in DCD Subsection 4.5.1.1.

GE Response

Please see response to RAI 4.5-13.

NRC RAI 4.5-29

DCD Section 4.5.1.2.1 states that the degree of conformance to RG 1.44 is presented in Subsection 4.5.2.4. Subsection 4.5.2.4 states that "These controls are employed to comply with the intent of RG 1.44." The word "intent" does not make it clear if the application meets the RG positions. If the ESBWR design does not meet all of the provisions of RG 1.44, please list the deviations and provide a basis. If the ESBWR design does meet the RG positions, please correct the language in Subsection 4.5.2.4 and any other applicable Subsection of the DCD to reflect that the design complies with RG 1.44. Specify the test used to comply with the guidance provided in RG 1.44. Provide the response in a global context as it applies to the entire ESBWR design.

GE Response

ESBWR stainless steel core support structures and internal components will comply with RG 1.44. The use of the word "intent" was in the general sense that the intent of RG 1.44 is that sensitized stainless steel will not be used. ESBWR complies with this intent by using all low carbon materials that have not been subjected to furnace sensitization and, because they are low carbon, are not sensitized by welding or other fabrication processes. All stainless steel materials used to fabricate ESBWR core support structures and internal components will be subjected to rigorous sensitization testing. This test is a modified version of ASTM A 262 Practice A wherein the definition of rejectable ditching is more strictly defined than the ASTM version, and retest and acceptance by Practice E is not allowed.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-30

DCD Subsection 4.5.1.4 references NQA-1, part 2.2 and RG 1.37. Please specify the edition of NQA-1 that is applicable. The staff notes that the ESBWR DCD references NQA-1-1983 in Chapter 17 but the applicable section related to requirements for cleaning of fluid systems and associated components is located in NQA-2-1983. Please provide clarification and state if all positions of RG 1.37 are being met. Provide the response in a global context as it applies to the entire ESBWR design.

GE Response

The ESBWR design commitment in DCD Table 1.9-22 will be changed to NQA-1-1983 and NQA-2-1983 in response to NRC review of DCD Chapter 17. All references in the DCD to NQA-1 and/or NQA-2 will be revised accordingly.

The ESBWR design complies with RG 1.37 except as noted in DCD Table 1.9-21b. The NRC has accepted an alternate position as documented in Table 2-1 of DCD Reference 1.9-2 (GE Nuclear Energy Quality Assurance Program Description, March 31, 1989, NEDO-11209-04a, Class I (non-proprietary) Revision 8). The alternate position is stated as follows:

"Comply with the provisions of Regulatory Guide 1.37, March 16, 1973, including the requirements and recommendations in ANSI N45.2.1-1973, except as follows: Section 5, sixth paragraph, recommends that local rusting on corrosion resistant alloys be removed by mechanical methods. This recommendation shall be interpreted to mean that local rusting may be removed mechanically, but that it does not preclude the use of other removal means."

In addition, the ESBWR design complies with the cleaning requirements of ANSI N45.2.1-1980 and the packaging, shipping, receiving, storage and handling requirements of ANSI N45-2.2-1978 as referenced in DCD Table 1.9-22. Compliance is met by means of their incorporation into NQA-2-1983.

DCD Section 4.5 will be revised in the next update to specify "NQA-2-1983, Part 2.2" in Subsection 4.5.1.4 instead of "NQA-1, Part 2.2".

NRC RAI 4.5-31

DCD Section 4.5.1.1 states that for incidental cold work introduced during fabrication and installation, special controls are used to limit the induced strain and hardness, and bend radii are kept above a minimum value. Please provide the values of the ESBWR design special controls limits on hardness, 0.2% offset yield strength and induced strain. Also discuss abrasive work controls for limiting cold working and the introduction of contaminants during abrasive work. Provide the response in a global context as it applies to the entire ESBWR design.

GE Response

GE applies special cold work controls to all stainless steel in the reactor system, defined as components inside containment continuously exposed to reactor water greater than 93°C (200°F). Bulk hardness of all stainless steels in the final fabricated condition (with the one exception noted in the response to RAI 4.5-12) is controlled to Rockwell B-90 for Types 304/304L and Rockwell B-92 for Types 316/316L. Cold forming and straightening strains are limited to 2.5%, or alternately, in the case of bars, plate, or pipe, a bend radius greater than 20 d or t (diameter or thickness). Additionally, for the major structural welds of core support structures and large internal components, polishing of the weld heat affected zones is required to remove surface cold work introduced by forming, machining, or grinding. Maximum yield strength is not controlled specifically, but the combination of solution heat treatment controls, hardness controls, and cold forming controls assures that, in all cases, the yield strength of stainless steels is far below 90,000 psi. Grinding is controlled by requiring ground areas to be polished to remove surface cold work introduced by grinding. Grinding media are controlled by requirements that processing materials shall be low in halogens, sulfur, and low melting point metals as well as thorough final cleaning of all ground surfaces. Additionally, it is required that grinding media be new, or previously used only on stainless steel or nickel alloys.

No DCD changes will be made in response to this RAI.

NRC RAI 4.5-32

DCD Subsection 4.5.2.4 describes delta ferrite content for weld materials and specifies the minimum and maximum values for the ESBWR design but the application, as it relates to Subsection 4.5.1, does not state if the ESBWR design meets all of the provisions of RG 1.31. If the ESBWR design does not meet all of the provisions of RG 1.31, please list the deviations and provide a basis. If the ESBWR design does meet the RG positions, please update Subsection 4.5.2.4 and any other applicable Subsection of the DCD to reflect that the design complies with RG 1.31.

GE Response

Stainless steel weld metals used for ESBWR core support structures and internal components will comply completely with RG 1.31 and exceed the provisions of the Regulatory Guide in several aspects. Ferrite control as described in DCD Subsection 4.5.2.4 is applied to all stainless steel weld filler regardless of application, including cladding. Magnetic measurement of an undiluted weld pad, as described in ASME NB-2400 is required. Magnetic instruments are restricted to Magnegage and Ferritescope calibrated according to AWS A4.2. These requirements are applied to all internal components, not just core support structures, and with no exemptions based on heat treatment condition or use of consumable inserts.

DCD Subsection 4.5.2.4 will be revised in the next update to indicate that ESBWR core support structures and internal structures will fully comply with RG 1.31. The relevant paragraph will be modified as follows:

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel—Regulatory Guide Conformance

Significantly cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers. Applying limits on hardness controls cold work, bend radii and surface finish on ground surfaces. Furnace sensitized material are not allowed. Electroslag welding is not applied for structural welds. *ESBWR will comply fully with Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal" including application of the following provisions to all stainless steel weld filler metal applied to reactor internal components:* The delta ferrite content for weld materials used in welding austenitic stainless steel assemblies is verified on undiluted weld deposits for each heat or lot of filler metal and electrodes. The delta ferrite content is defined for weld materials as 5.0 Ferrite Number (FN) minimum, 8.0FN average and 20FN maximum. *Ferrite content will be determined by use of magnetic instruments calibrated according to AWS A4.2.* This ferrite content is considered adequate to prevent any micro-fissuring (Hot Cracking) in austenitic stainless steel welds. This procedure complies with the requirements of Regulatory Guide 1.31.

Enclosure 2

MFN 06-178

Response to NRC Request for Additional Information Letter

No. 29 Related to ESBWR Design Certification Application

Integrity of Reactor Coolant Pressure Boundary

RAI Numbers 5.2-6 through 5.2-29

NRC RAI 5.2-6

DCD Tier 2, Section 5.2.2 lists the applicable GDC. In addition to GDC 15, the overpressure protection and pressure control devices inboard of the main steam isolation valves (MSIVs) are considered part of the reactor coolant pressure boundary (RCPB), therefore, GDC 14 and 30 are applicable. Provide a discussion of how GDC 14 and 30 are met.

GE Response

GDC 14 is met by providing SRVs to protect the RCPB from overpressure. Additionally, the SRVs and associated piping are designed, fabricated, erected and tested in accordance with 10 CFR 50.55a to provide a high degree of integrity. The SRVs are classified as Quality Group A. The fracture toughness properties and operating temperature of ferritic materials are controlled to ensure adequate toughness. Material and examination requirements are established for the SRVs and their associated piping prior to and after their assembly and erection. GDC 30 is met by the use of conservative design practices and detailed quality control procedures during the design and fabrication of the SRVs and their associated piping so that integrity is retained during normal and postulated accident conditions. For additional discussion, please see Chapter 3, Design of Structures, Components, Equipment and Systems, Section 3.1.2.5 for GDC 14 and 3.1.4.1 for GDC 30. DCD Section 5.2.2 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-7

DCD Tier 2, Section 5.2.2 states that "The ESBWR meets the recommendations of the TMI action plan items of NUREG 0737 regarding testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents and the provision of direct indication of relief and safety valve position."

TMI-2 action item II.D.1 requires licensees to provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves for all fluid conditions expected under operating conditions, transients and accidents. Consideration of ATWS conditions shall be included in the test program. For current operating plants, this issue was resolved with a generic test program for current valve designs and plant-specific responses for individual plant piping configurations and system responses. Confirm that the generic test program for the currently operating plants is applicable for ESBWR transients, ATWS, and accidents, or provide a commitment to perform the required testing. Also, provide a commitment to provide necessary plant specific responses as a combined operating license (COL) action item.

GE Response

ESBWR design meets the recommendations of TMI action plan item II.D.1 in 10 CFR 50.34(f)(2)(x) regarding a test program and associated model development and testing to qualify reactor coolant system relief and safety valves for all fluid conditions expected under operating conditions, transients and accidents. Consideration of ATWS conditions is included in the test program. Consistent with past practice, a purchase specification for the SRV will be prepared that addresses the inspection and test requirements of the test program. The testing and inspection of the SRVs utilizes a quality assurance program that complies with Appendix B of 10 CFR 50. The tests include hydrostatic, steam leakage, full flow pressure and blowdown and response time testing. Please see DCD Tier 2, Table 1A-1 for additional information. During plant startup testing, the SRVs are tested to confirm adequate flow path from inlet to outlet, operability of the valve and response during transient testing. The SRV plant specific responses are obtained after plant startup and, therefore, are beyond the boundaries of the COL process. Therefore, there is no applicable COL action item. DCD Section 5.2.2 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-8

In accordance with the requirements of TMI-2 action item II.D.3, safety relief valve (SRV) position indications are provided in the control room. In some operating plants, thermocouples and acoustic monitors are provided at the SRV discharge piping for redundancy and diversity. Why are these diverse and redundant features not included in the ESBWR design?

GE Response

ESBWR design meets the recommendations of TMI action plan item II.D.3 in 10 CFR 50.34(f)(2)(xi) regarding SRV position indications in the control room. Open and closed indications for the SRVs are provided in the control room. This is addressed in DCD Tier 2, Table 1A-1. Additionally, there are thermocouples installed in the SRV discharge piping that provide redundancy in detecting open and closed position of the SRV. This is consistent with current operating BWRs and the ABWR design. DCD Section 5.2.2 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-9

Address TMI-2 action item II.K.3.3 in regards to reporting SRV challenges and failures. This may be a COL action item which is to be tracked.

GE Response

SRV OPERABILITY requirements are included in the Technical Specifications (TS) provided in DCD Chapter 16 (LCO 3.4.1). Licensees are expected to report SRV challenges and failures in accordance with the requirements of 10 CFR 50.73, "Licensee event reports." Therefore, consistent with Federal Register Notice 69 FR 35067 on June 23, 2004, "Notice of Availability of Model Application Concerning Technical Specifications Improvement To Eliminate Requirements to Provide Monthly Operating Reports and Occupational Radiation Exposure Reports Using the Consolidated Line Item Improvement Process" and with Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-258 referenced therein, the ESBWR TS do not include the additional monthly reporting requirement referenced in Section III.6.d of SRP 5.2.2 DRAFT Rev. 3 - April 1996 and a COL action item is not needed for this issue.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-10

Confirm that American Society of Mechanical Engineers (ASME) Code NB-7623 provisions will be met for rupture disc installation. Provide details of the specific SRV design features which ensure that the rupture disc will not impede SRV response time or operation.

GE Response

The design of the rupture disk will comply with ASME Code NB-7623. The rupture disk will be designed with a burst pressure high enough so that it remains closed if an SRV is in a simmering condition and leaking steam, but low enough so that it will open at a pressure that does not adversely affect the opening of the SRV. Direct acting SRVs are used extensively in current operating BWRs and in the ABWR design. Based upon past practice, the SRV discharge piping and components are sized so that critical flow conditions occur through the valve. This ensures that backpressure is maintained within limits and that the operation and response time of the SRV are not adversely affected. This is addressed in the design of the ESBWR SRV and rupture disk.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-11

It *is stated in DCD Tier 2, Section 5.2.2.2.2 on page 5.2-5, that "[e]ight of the SRVs are opened by steam pressure initiated if the direct and increasing static inlet pressure overcomes the restraining spring and the frictional forces acting against the steam inlet pressure at the main or pilot disk and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures." Essentially identical language is used to describe the 10 SRV/automatic depressurization valves (ADS). This statement is confusing. Should the statement read: "the main (or pilot) disk opens quickly when the steam inlet pressure exceeds the restraining spring force and frictional forces"?*

GE Response

All eighteen SRVs are capable of opening in response to sufficiently high inlet steam pressure. This will be clarified in DCD Section 5.2.2.2. Additionally, the statements describing the safety (steam pressure) mode of operation will be by revised as suggested to eliminate the confusion. DCD Section 5.2.2.2.2 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-12

Operating experience shows that there have been events involving failure of the SRV discharge line vacuum breakers to open or close properly. Some events have occurred as a result of condensation pressure oscillations which resulted from routine SRV seat leakage. Provide an analysis of the failure of a vacuum breaker to either open or close properly during or following SRV discharge to demonstrate that discharge lines remain operable and that reactor coolant system and containment design criteria are met.

GE Response

There have been few reported instances of vacuum breaker failure. However, ESBWR vacuum breaker design will incorporate the most recent design experience with the Lungmen ABWR design in order to minimize any chance of failure. In the event that a vacuum breaker opens, the capability of the SRV to perform its safety function is evaluated. According to a review of SRV history at current operating BWRs, no cases have been identified of SRV leakage affecting the vacuum breakers. The few identified vacuum breaker anomalies have been associated with an SRV opening and then re-closing. If a vacuum breaker opens and sticks open, then upon subsequent SRV actuation the steam will discharge inside the Drywell instead of to the Suppression Pool. Over time, the discharged steam inside the Drywell will produce a high Drywell pressure and/or temperature, which will ultimately produce a scram to safely shutdown the plant. However, most likely the steam discharge pressure from SRV actuation will force the vacuum breaker closed, which allows steam to be discharged to the SP. If a closed vacuum breaker is stuck in the closed position, then upon SRV returning to its closed position after actuation, the water level will tend to rise in the downcomer piping. However, the water rise will be prevented since there is a second vacuum breaker in parallel with the failed vacuum breaker. It is very unlikely that two vacuum breakers in parallel will fail. Therefore, there is no adverse effect produced by a failed vacuum breaker on the function of the SRV.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-13

In some currently operating boiling water reactors, there has been excessive corrosion bonding of SRV pilot disks to their seats, causing excessively high set-point drift. A contributing cause was found to be the presence of radiolytic oxygen buildup, and one corrective action was to recombine the oxygen with hydrogen using a catalyst to form water. Aside from maintaining oxygen concentration through hydrogen addition, what other provisions will be made to prevent excessive set-point drift from corrosion bonding in SRVs for the ESBWR?

GE Response

This problem has been associated with pilot operated type SRVs. Direct acting type SRVs are used extensively in current operating BWRs and the ABWR design. There is no history of this problem with direct acting type SRVs. Selection of the ESBWR SRV design has not been finalized. However, because there is no history of this problem with direct acting type SRVs and with consideration of lessons learned over many years of experience, the most likely choice for the ESBWR design is a direct acting type SRV. Additionally, in-service and surveillance tests are performed to confirm the operability of the SRVs. Therefore, it is not expected that this problem will occur with the ESBWR design SRVs. However, if a pilot operated type valve is considered for the ESBWR design, the likelihood of sticking and bonding will be rigorously reviewed and evaluated.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-14

DPVs and non-ADS SRVs are discharged into the drywell. Is there any safety grade equipment in the drywell which is required for safe shutdown of the plant which will be affected by the DPV and SRV discharge into the drywell?

GE Response

The location of all equipment in the Drywell has not been finalized at this time. However, according to the current arrangement of major equipment in the Drywell, there is no safety grade equipment required for safe shutdown of the plant that is affected by DPV or Non-ADS SRV discharge. As the process continues to locate all equipment, safety grade equipment required for safe shutdown of the plant will not be located in a position that is affected by DPV or non-ADS SRV discharge in the Drywell. Clarification will be provided on DCD Figure 5.2-1 to show that Non-ADS SRVs discharge to the Drywell. DCD Figure 5.2-1 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-15

DPV capacities are given in DCD Tier 2, Table 5.2-2. It is not clear whether credit is taken for the DPVs in the overpressure analysis. Confirm that credit is not taken for DPVs in the overpressure analysis.

GE Response

DPVs do not mitigate the over pressure event. A clarifying note will be added to DCD Table 5.2-2. DCD Table 5.2-2 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-16

Operational history for boiling water reactors indicates that SRV liquid discharge may occur as a result of reactor vessel overfilling into the main steam lines. Provide information regarding SRV liquid discharge in the event of vessel overfilling to demonstrate that the SRVs remain operable and that the SRV discharge lines and main steam lines do not exceed the applicable stress limits.

GE Response

Low-pressure water flow requirements are typically included in the SRV purchase specification. Additionally, SRV discharge line loads (e.g. chugging) are Nuclear Boiler System and SRV design considerations. During the post-TMI period, all BWR SRVs were tested with low-pressure water with no adverse effects. Therefore, in the event of vessel overfilling under low-pressure conditions in the ESBWR design, any adverse effect on the function of the SRV is not expected. During power operation and high pressure conditions it is very unlikely that the main steam lines will be filled with water because more than one failure must occur, such as failure of the high RPV level scram, failure of the feedwater control system, failure of the runback of feedwater demand to zero and failure of the trip of the feedwater pumps. However, several years ago at an operating BWR there was a case of overfilling of the main steam lines at high-pressure conditions. Subsequently, the SRVs were opened and water was discharged to the suppression pool. It was observed that the SRVs opened slower than expected. The SRVs were removed and inspected. There was no identified damage to the SRVs. The SRVs were refurbished and recertification tested and returned to service. Discharge line piping and hangers were also inspected. There were no apparent problems. Post TMI testing showed that more time is required for the SRV to respond to an open command and reach full open under water conditions as compared to steam. Therefore, it has been demonstrated that the SRVs and associated piping remain operable under low-pressure and high-pressure water conditions. These lessons learned will be considered during the process of evaluation and selection of the ESBWR SRV.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-17

DCD Tier 2, Section 5.2.2.3.2, states that "SRV capacities are based on establishing an adequate margin from the peak vessel bottom pressure to the vessel code limit." Explicitly define "adequate margin." If the margin includes any factors besides the 3% SRV set-point pressure tolerance, describe these additional factors.

GE Response

Overpressure protection margin is one of the features, which differentiates ESBWR from operating BWRs. The analysis of the pressure rise for the ESBWR design shows that the peak vessel bottom head pressure is 8.71 MPa gauge. The vessel code limit is 9.481 MPa gauge. Therefore, the margin between the peak vessel bottom head pressure and the vessel code limit is 8.8%. According to the ASME Code, there is no specifically defined margin. Therefore, the margin of 8.8% is considered adequate. The number of installed SRVs provides additional margin. According to the response for RAI 5.2-27, the capacity of one SRV is adequate to pass enough steam to prevent overpressurization. There are 18 SRVs used in the ESBWR design. The additional 17 SRVs are required to handle the ATWS conditions and, therefore, provide additional margin for overpressure protection.

Additionally, in response to a request during the telecon between the NRC and GE on 4/25/06, it should be noted that Table 5.2-2 addresses the requirement for the SRVs to provide rated flow at a pressure that does not exceed the opening actuation pressure by more than 3%. The 3% is based upon the ASME Code Section III, NB-7532.3.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-18

Specify the version of the TRACG code used for the overpressure protection analysis, and provide the appropriate reference in DCD Tier 2, Section 5.2.7

GE Response

TRACG04 is the version used for the overpressure protection analysis. DCD Section 5.2.7 will be revised in the next update to include the appropriate reference as noted below:

5.2-9 General Electric Company, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis" NEDE-32906P, Revision 1, April 2003.

NRC RAI 5.2-19

Why is the automatic power-actuated pressure relief function not included in the ESBWR design?

GE Response

The ESBWR pressurization is mild relative to previous BWR designs because of the large steam volume in the chimney and vessel head, which mitigates the pressurization during the most severe transient and does not result in opening of relief valves prior to isolation condenser system initiation. The SRVs are set high enough such that the only time an SRV will open, other than an event requiring ADS operation, is during an ATWS event. Eliminating the power-actuated pressure relief function eliminates a potential cause of inadvertent opening because of a spurious high-pressure signal. This satisfies the requirements in the URD of minimizing the chance of inadvertent SRV opening.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-20

SRV set point drift and seat leakage are generic problems. Describe specific design features of the ESBWR SRVs. Compare the relative performance of ESBWR SRVs and SRVs currently installed in operating reactors. Provide a detailed description of any improvements between ESBWR SRV designs and presently operating plant SRVs in the areas of seat-leakage, set-point drift, and actuator reliability.

GE Response

The detailed design and selection of the ESBWR SRVs have not been finalized. Lessons learned from valves installed in current operating BWRs will be considered during the selection phase of the ESBWR SRVs. Historically, set point drift has been a concern with pilot operated valves. However, this concern has not been a problem with a direct acting type valve, which is in use in many of the more recent design BWRs. Seat leakage has been a concern with some direct acting valves. However, valve suppliers have continued to evaluate and incorporate design modifications pertaining to seat and disc geometry and materials to reduce seat leakage. GE will evaluate potential valve suppliers during the selection phase with an emphasis on optimum performance in the areas of set point drift, actuator reliability and seat leakage. Selection of a direct acting SRV for ESBWR is the most likely choice and provides an improvement over SRVs in current operating BWRs, which use 18 SRVs that are capable of remote actuation. The ESBWR design has reduced the number of SRVs that are capable of remote actuation from 18 to 10 because fewer SRVs are needed for depressurization because of the addition of the DPVs. This reduces the chance of inadvertent actuation and subsequent seat leakage. The remainder of the SRVs in the ESBWR are designed for safety mode only and are not capable of remote actuation. Both groups of valves are designed with nameplate set pressures that are higher than the set pressure of SRVs in current operating BWRs. This produces a higher simmer margin and makes it less likely that there will be leakage through the SRV. DCD Section 5.2.2.2.2 will be revised to clarify that only 10 SRVs are remote actuated from the control room.

The operability of the SRV when discharging water under low-pressure and high-pressure conditions was addressed in response to RAI 5.2-16. Review of the occurrence under high-pressure conditions showed that there was no water hammer, but there was a high intial SRV discharge line loading when the SRV first opened. Subsequent investigation showed that there were no adverse effects on the SRV and its associated piping. Therefore, water hammer is not expected to be a problem with the ESBWR SRVs.

DCD Section 5.2.2.2.2 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-21

Include a discussion of improvements in the air actuator, especially materials used for components such as diaphragms and seals. Discuss the safety margins associated with the air accumulator design. Discuss the pressure indications in the accumulator and how this information is relayed to the operator.

GE Response

Improvements in Air Actuator: From an environmental standpoint, the limiting actuator materials are those used for elastomer type seals. As part of NUREG-0588 qualification programs, changes have been made in actuator seal materials that provide greater environmental capability. These material improvements are incorporated into the latest SRV designs and demonstrated capable by type testing.

Safety Margins: Accumulator volume is sized based upon actuation at Drywell design pressure. Accumulator is sized to provide sufficient pressure to open the SRV assuming the actuating pressure is at its minimum value and the pneumatic supply to the accumulator has failed.

Pressure Indication: No pressure indication is available at the accumulator, which is consistent with current operating BWRs and the ABWR design. The High Pressure Nitrogen Supply System (HPNSS) supplies nitrogen to the accumulators. There is pressure indication on the header in the HPNSS. This indication is supplied to the control room to alert the operators of a degradation in the supplied pressure.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-22

What provisions have been employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (i.e. radiation, temperature, humidity, and vibration)?

GE Response

A purchase specification, which utilizes the extensive GE Environmental Qualification experience base, will be prepared for the SRV. This specification defines the design and qualification requirements for the SRV. The SRV will be subjected to Environmental and Dynamic Qualification as defined in the purchase specification. The purchase specification will define the environmental conditions such as radiation, temperature, pressure and humidity and the seismic and dynamic conditions, which include the required response spectra. Also, the purchase specification will define the requirements for the Environmental and Dynamic Qualification Program, such as radiation aging, thermal aging, mechanical aging and vibration testing.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-23

In DCD Tier 2, Section 5.2.2.5, why are SRV position indication and seat-leakage detection not instrumentation requirements for the ESBWR?

GE Response

Position indication and seat-leakage detection are SRV instrumentation requirements and, therefore, will be added to DCD Section 5.2.2.5. DCD Section 5.2.2.5 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-24

What programs have been instituted to ensure that valves are manufactured to specifications and will operate as designed? For example, what tests are performed to ensure that the blowdown capacity is within specifications?

GE Response

Consistent with past practice, a purchase specification will be prepared for the SRV. This specification defines the design requirements, fabrication requirements, materials requirements, inspection and testing requirements, cleaning and packaging requirements and document submittal requirements. Review of document submittals and surveillance inspections will confirm that the SRV is manufactured to specifications. Design Confirmation testing is performed to confirm the adequacy of the SRV design and that the operation of the SRV complies with the design requirements. The requirements for flow capacity are defined in the purchase specification. Flow capacity testing is performed for the SRV in a full flow test loop. This provides assurance that the SRV set pressure and reseat/blowdown criteria are met. The testing also demonstrates that the flow capacity rating is met and certified in accordance with the ASME Code.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-25

Operating experience has shown that SRV failure may be caused by exceeding the manufacturer's recommended service life for the internals of the SRV or air actuator. What frequency is necessary for visual inspection and overhaul of the SRVs (both safety/relief and ADS valve components)? In addition to periodic testing, valve inspection and overhaul should be performed in accordance with the manufacturer's recommendations. Provide assurance that procedures will be in place to ensure that the design service life will not be exceeded for any component of the SRV?

GE Response

The SRVs are mounted on flanges and, therefore, can be removed for maintenance or bench testing during normal plant shutdown. The valves are tested in accordance with the in-service testing program as discussed in DCD Tier 2, subsection 3.9.6 and Table 3.9-8. Every 5 years during reactor plant shutdown, the valves are subjected to a complete visual examination, set pressure testing and seat tightness testing. External and flange seating surfaces of the SRVs are 100% inspected when any valve is removed for maintenance or bench testing. At every refueling outage, valve position verification and exercising tests are performed for the SRVs. Based upon past practice with SRVs, the valve manufacturer provides an equipment instruction manual, which provides maintenance recommendations and instructions for servicing and overhaul of valve components and parts. The instruction manual is a committed deliverable to the plant owner and, therefore, it's contents are incorporated into the plant maintenance staff's maintenance procedures, which ensure that the design life will not be exceeded for any component of the SRV.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-26

DCD Tier 2, Section 5.2.2.2.2 states: "The power-actuated SRVs can be operated individually by remote manual controls from the main control room." This implies that only the ten ADS valves can be opened from control room. Can the other eight non-ADS valves be remotely opened and closed? 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:

Only the ten ADS-SRVs can be operated from the main control room. When the ADS-SRVs are actuated in response to ADS initiation, they open and remain open and cannot be closed by the operators because the ADS initiation signal is sealed in. The eight Non-ADS SRVs cannot be remotely opened and closed.

No DCD changes will be made in response to this RAI.

NRC RAI 5.2-27

Provide a figure showing the peak vessel bottom pressure as a function of the number of SRVs assumed to open in the isolation overpressure analysis. Identify the minimum number of SRVs that must open to remain within ASME limits on vessel pressure.

GE Response

Figure 5.2-4b shows that the steam flow required to prevent continued pressurization is about 2% of rated steam flow (sustained flowrate after SRV opening). Each SRV provides 5% of rated steam flow. Using the information on the figure, it can be concluded that only one SRV is needed to prevent over-pressurization. Figure 5.2-4b shows that after the SRVs open there are 0.5 full power seconds of steam flow before the flow rate settles at 2%, which occurs within 10 seconds. The time required for one SRV to pass the flow needed to stop the pressurization is calculated to be 30 seconds. From Figure 5.2-4d, when the bottom head pressure curve is extrapolated from time 38 seconds at the same pressurization rate, i.e. same slope, for 20 seconds the resulting pressure is 9 MPa and is within the ASME limit. Therefore, only one SRV is needed to prevent exceeding the ASME limit in the ASME overpressure protection event. The ESBWR design includes 18 SRVs. The other 17 SRVs are needed for the ATWS event. The following will be added to the last paragraph of DCD Section 5.2.2.3.2:

"Only one SRV is required to open to prevent exceeding the ASME limit in the ASME overpressure protection event. Eighteen SRVs are included in the ESBWR design. The other 17 SRVs are needed for the ATWS event."

NRC RAI 5.2-28

DCD Tier 1, Table 2.1.2-1 does not include spring set-points as part of the Inspections, Tests, Analysis and Acceptance Criteria (ITAAC) for the SRVs. Add the spring set pressures for all the SRVs to the ITAAC.

GE Response

Will add SRV spring setpoints to ITAAC. DCD Table 2.1.2-1 and ITAAC Table 2.1.2-2, item #10 will be revised in the next update as noted in the attached markup.

NRC RAI 5.2-29

The following ITAAC were included in the ABWR DCD. Explain why similar ITAAC are not provided the ESBWR DCD.

(a) The ADS can be initiated manually.

(b) The reactor pressure vessel (RPV) water level instrumentation considers the effects of dissolved noncondensable gasses in the RPV water instrument lines.

(c) The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building is physically separated from other divisions.

(d) Motor operated valves designated in Section 2.1.2 as having an active safety function will close under differential pressure, fluid flow, and temperature conditions.

(e) Control valves designated in Section 2.1.2 as having an active safetyrelated function will actuate (open, close, or both open and close), under differential system pressure, fluid flow, and temperature conditions

(f) The ADS accumulator can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator

GE Response

These ITAACs will be added. DCD ITAAC Table 2.1.2-2 will be revised in the next update as noted in the attached markup.

5.2.2 Overpressure Protection

This subsection evaluates systems that protect the RCPB from overpressurization.

As noted in SRP 5.2.2 Draft R3, overpressure protection for the reactor coolant pressure boundary (RCPB), during power operation of the reactor, is in compliance with ASME B&PV Code Section III and is ensured by application of relief and safety valves and the reactor protection system. For the ESBWR, the equipment includes Safety-Relief Valves (SRVs) on the main steam lines and piping from the SRVs to the suppression pool.

Overpressure protection for the RCPB, during low temperature operation of the plant (startup, shutdown), is ensured by the application of pressure relieving systems that function during the low temperature operation. For BWRs, no special area of review is required because BWRs never operate in water-solid conditions.

The ESBWR overpressure protection system meets the relevant requirements of the following regulations:

(1) General Design Criterion 15, as it relates to the reactor coolant system and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

(2) General Design Criterion 14, as it relates to the reactor coolant pressure boundary being designed, fabricated of, erected and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure and of gross rupture.

(3) General Design Criterion 30, as it relates to components, which are part of the reactor coolant pressure boundary, being designed, fabricated, erected and tested to the highest quality standards practical.

The ESBWR design meets the recommendations of the TMI action plan item~~s~~ II.D.1 in 10 CFR 50.34(f)(2)(x) regarding a test program and associated model development and testing~~ of NUREG-0737 regarding testing~~ to qualify reactor coolant system relief and safety valves for all fluid conditions expected under ~~expected~~ operating conditions, ~~for~~ design basis transients and accidents. The ESBWR design also meets the recommendations of TMI action plan item II.D.3 in 10 CFR 50.34(f)(2)(xi) ~~and the provision of~~regarding SRV position indication by providing open and closed ~~direct~~ indication of ~~relief and safety~~each valve ~~position~~in the control room.

Other specific acceptance criteria of GDC 15 met by ESBWR are as follows:

For overpressure protection, the Isolation Condensers have sufficient capacity to preclude actuation of the SRVs, during normal operational transients, when assuming the following conditions at the plant:

a. The reactor is operating at licensed core thermal power level.

b. All system and core parameters are at values within normal operating range that produce the highest anticipated pressure.

c. All components, instrumentation, and controls function normally.

5.2.2.2.2 Equipment and Component Description

Description

The nuclear pressure-relief system consists of 18 SRVs located on the main steamlines between the reactor vessel and the first isolation valve within the drywell. The SRVs are flange mounted onto forged outlet fittings located on the top of the main steamline piping in the drywell. Ten SRVs are designated as ADS SRVs and each discharges through a lines routed to a quenchers in the suppression pool. The remaining eight SRVs are designated as Non-ADS SRVs and are arranged into two groups of four. Each group discharges to a horizontal header that has a rupture disc at each end. Each header has a discharge line that is routed to a quencher in the suppression pool. These Non-ADS SRVs discharge through the rupture discs to the drywell or through the discharge line to the suppression pool. These SRVs ~~valves~~ protect against overpressure of the nuclear system and allow for manual or automatic reactor system depressurization.

The SRVs provide two main protection functions:

- overpressure safety operation (all eighteen of the valves are actuated by inlet steam pressure to prevent nuclear system overpressurization);
- depressurization operation [ten of the valves are actuated by the Automatic Depressurization System (ADS) as part of the Emergency Core Cooling System (ECCS) for events involving breaks in the nuclear system process barrier]

Chapter 15 discusses the events that are expected to activate the primary system SRVs. It also summarizes the number of valves expected to operate in safety (steam pressure) mode of operation during the initial blowdown of the valves and the expected duration of this first blowdown. In response to an event that activates the SRVs, remote manual actuation of the SRVs from the control room is recommended to minimize the total number of these discharges with the intent of achieving extended valve seat life.

~~Remote manual actuation of the SRVs from the control room is recommended to minimize the total number of these discharges with the intent of achieving extended valve seat life.~~

All eighteen ~~Eight~~ of the SRVs are opened by the safety (steam pressure) mode of operation. These SRVs open by steam pressure when ~~are opened by steam pressure initiated if the direct and increasing static inlet steam pressure overcomes the restraining spring and the frictional forces acting against the inlet steam pressure at~~ the main (or pilot) disk opens quickly in response to the steam inlet pressure exceeding the restraining spring force and frictional forces~~and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures.~~ The condition at which this action is initiated is termed the "popping pressure" and corresponds to the set-pressure value stamped on the nameplate of the SRV.

In addition to the safety (steam pressure) mode of operation, the ~~Ten~~ ten ~~of the~~ADS SRVs are opened by ~~either of~~ the following ~~two modes~~ mode of operation:

- ~~The safety (steam pressure) mode of operation is initiated when the direct and increasing static inlet steam pressure overcomes the restraining spring and the frictional forces acting against the inlet steam pressure at the main or pilot disk~~

~~and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures. The condition at which this action is initiated is termed the "popping pressure" and corresponds to the set pressure value stamped on the nameplate of the SRV.~~

- The ADS (power) mode of operation is initiated when an electrical signal is received at any of the solenoid valves located on the pneumatic actuator assembly. The solenoid valve(s) open, allowing pressurized nitrogen to enter the lower side of the pneumatic cylinder piston, which pushes the piston and the rod upwards. This action pulls the lifting mechanism of the main or pilot disk, thereby opening the valve to allow inlet steam to discharge through the SRV until the inlet pressure is near or equal to zero or the solenoid valve is closed.

The pneumatic operator is so arranged that, if it malfunctions, it does not prevent the valve from opening when steam inlet pressure reaches the spring lift set pressure.

For overpressure SRV operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at the setpoint designated in Table 5.2-2. In accordance with the ASME Code, the full lift of this mode of operation is attained at a pressure no greater than 3% above the setpoint. The opening time for the SRVs, from the time the pressure exceeds the valve set pressure to the time the valve is fully open, is less than 1.7 second.

The ten ADS ~~power-actuated~~ SRVs are power-actuated and can be operated individually by remote manual controls from the main control room.The eight Non-ADS SRVs are not capable of remote actuation from the main control room.

The ADS ~~utilizes ten of the~~ SRVs are utilized for depressurization of the reactor as described in Section 6.3. Each of the SRVs is equipped with a pneumatic accumulator and check valve for the ADS and the manual opening functions. These accumulators assure that the valves can be opened following failure of the gas supply to the accumulators. The accumulator capacity is sufficient for one actuation at drywell design pressure.

Each ADS SRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool water level. The discharge lines enter the suppression chamber below the suppression pool water level. The discharge lines are classified as Quality Group C and Seismic Category I.

Two vacuum relief valves are provided on each SRV discharge line to minimize initial rise of water in discharge piping and prevent drawing an excessive amount of water into the line as a result of steam condensation following termination of relief operation.

The ADS, which consists of SRVs and DPVs, automatically depressurizes the nuclear system sufficiently to permit the GDCS to operate. Further descriptions of the operation of the automatic depressurization feature are presented in Section 6.3.2.8.2 and within Subsection 7.3.1.

Design Parameters

The specified operating transients for components within the RCPB are presented in Section 3.9. Subsection 3.7.1 provides a discussion of the input criteria for design of

Seismic Category I structures, systems, and components. The design requirements established to protect the principal components of the reactor coolant system against environmental effects are presented in Section 3.11.

Safety-Relief Valve

The design pressure and temperature of the valve inlet is 9.48 MPa gauge (1375 psig) at 307°C (585°F).

The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology.

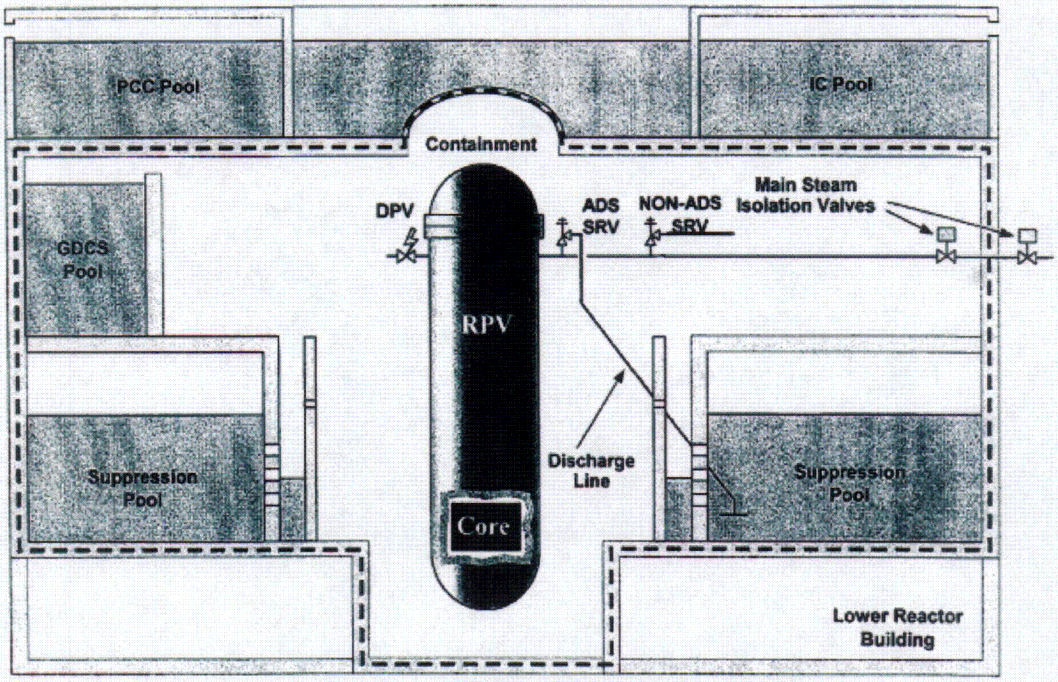


Figure 5.2-1. Safety/Relief Valve Schematic Elevation

Table 5.2-2

Safety/-Relief Valve and Depressurization Valve Settings and/or Capacities

Valve Type SRV /DPV	**Number of Valves**(1)	**Spring Setpoint Maximum Safety Analytical Limit** MPa gauge (psig)	**ASME Rated Capacity at 103% of Safety Analytical Limit Spring Setpoint Pressure** (kg/s each)
ADS SRV	10	8.618 (1250)	124
Non-ADS SRV	8	8.756 (1270)	126
DPV	8	NA	239 (2)

(1) The SRVs also perform the automatic depressurization function.

(2) Minimum capacity in ADS mode. The DPVs do not mitigate the overpressure event.

5.2.2.5 *Instrumentation Requirements*

~~None~~Each SRV discharge line contains a temperature element, which provides an indication of seat leakage within the valve or confirmation of valve opening. The temperature element provides a signal to an indicator and an alarm in the main control room. Each SRV has a position indicator, which provides a signal to the main control room for indication of open and closed position.

Table 2.1.2-1

SRV Capacities

Valves	Number of Valves	Spring Setpoint Maximum Safety Analytical Limit MPa gauge (psig)	ASME Rated Capacity at 103% Spring Set Pressure [(1)] kg/s (Mlb/hr) each	Used For ADS
Non-ADS SRV	8	8.618 (1250)	126 (1.000)	0
ADS-SRV	10	8.756 (1270)	124 (0.984)	10

DPV Capacities

Valves	Number of Valves	Capacity [(2)] kg/s (Mlb/hr) each	Used For ADS
DPV	8	239 (1.897)	8

(1) Minimum capacity per the ASME Boiler and Pressure Vessel Code, Section III.

(2) Minimum capacity in ADS mode.

Table 2.1.2-2

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the NBS is defined in Subsection 2.1.2.	1. Inspections of the as-built system will be conducted.	1. The as-built NBS conforms with the basic configuration as defined in Subsection 2.1.2.
2. Portions of the NBS are classified as ASME Code class as indicated in Subsection 2.1.2. They are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III.	2. ASME Code Data Reports will be reviewed and inspections of Code stamps will be conducted for ASME components in the NBS.	2. Those portions of the NBS identified as ASME Code Class in Subsection 2.1.2 have ASME Code Section III, Code Data Reports and Code stamps (or alternative markings permitted by the Code).
3. The throat diameter of each MSL flow limiter is less than or equal to 355 mm.	3. Inspections of the as-built MSL flow limiters will be taken.	3. The throat diameter of each MSL flow limiter is less than or equal to 355 mm.
4. Each MSL flow limiter has taps for two instrument lines. These instrument lines are used for monitoring the flow through each MSL.	4. Inspections will be conducted of the MSL instrument lines.	4. The MSL flow measurement instrument lines are installed.
5. The ASME Code portions of the NBS retain their integrity under internal pressures that will be experienced during service.	5. A hydrostatic test will be conducted on those Code components of the NBS required to be hydrostatically tested by the ASME Code.	5. The results of the hydrostatic test of the ASME Code components of the NBS conform with the requirements in the ASME Code, Section III.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The combined steamline volume from the RPV to the main steam turbine stop valves and steam bypass valves is greater than or equal to 135 m^3.	6. Calculations will be performed using the as-built dimensions of the steamlines to determine the combined steam line volume.	6. The combined steamline volume is greater than or equal to 135 m^3.
7. There are indications in the main control room for NBS parameters as defined in Subsection 2.1.2.	7. Inspections will be performed in the main control room of the NBS indications defined in Subsection 2.1.2.	7. The NBS indications defined in Subsection 2.1.2 are displayed in the main control room.
8. MSIV closing time is equal to or greater than 3 seconds and less than or equal to 5 seconds when N_2 or air is admitted into the valve pneumatic actuator. The MSIVs are capable of closing within 3 to 5 seconds under differential pressure, fluid flow and temperature conditions	8. Tests of the as-built MSIV will be conducted under preoperational test conditions. Tests or type tests, of an MSIV will be conducted under design basis differential pressure, flow and temperature conditions.	8. MSIV closing time is equal to or greater than [3 seconds] and less than or equal to [5 seconds].
9. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature of 20°C (68°F) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to or greater than 0.269 MPaD (40 psid).	9. Tests and analysis will be performed on the as-built MSIVs to determine the leakage.	9. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature of 20°C (68°F) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to or greater than 0.269 MPaD (40 psid).

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The SRV spring set pressures and flow capacities are given in Table 2.1.2-1. The opening time for the SRVs from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to 1.7 second.	10. Analysis and tests (at a test facility) will be conducted in accordance with the ASME Code.	10. Test reports and analyses exist and conclude that the SRVs have the capacities shown in Table 2.1.2-1. The opening time for the SRVs from when the pressure exceeds the valve set pressure to when the valve is fully open is less than or equal to 1.7 second.
11. The SRVs and DPVs are provided with instrumentation that will provide indication (i.e. by direct measurement) of valve position.	11. Inspection will be performed on the SRV and DPV position indication instrumentation.	11. The SRV and DPV position indicators provide open and close indication.
12. Upon receipt of an ADS initiation signal, the ADS logic generates signals to the SRVs and the DPVs.	12. Tests will be conducted using simulated input signals for each NBS process variable to cause trip conditions in the instrument channels of the same process variable associated with each of the ADS logic divisions.	12. Upon receipt of an ADS initiation signal, the ADS logic generates signals to the SRVs and the DPVs.
13. The 10 SRV discharge lines associated with the ADS function are piped directly to quenchers located below the surface of the suppression pool.	13. Inspections will be performed to review the configuration of the SRV discharge line quenchers.	13. The 10 SRV discharge lines associated with the ADS function have been installed and are piped directly to the quenchers located below the surface of the suppression pool.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14. When actuated by either of two initiators, the booster assembly opens the DPV in less than or equal to 0.45 seconds with an inlet pressure of 6.89 Mpa gauge (1000 psig.) or greater.	14. Tests will be performed on the booster assemblies during factory tests to confirm that they are capable of opening the valve. Tests and analyses will be performed to demonstrate that the booster opens the DPV.	14. Test reports and analyses exist and conclude that the DPV opens when actuated by the booster assembly in less than or equal to 0.45 seconds with an inlet pressure of 6.89 Mpa gauge (1000 psig.) or greater.
15. There are four DPVs attached to stub tubes off of the RPV and four DPVs attached to the main steam lines.	15. Inspections will be performed to review the configuration of the DPVs.	15. Four DPVs are attached to stub tubes off of the RPV and four DPVs are attached to the main steam lines.
16. The DPV minimum flow capacity is 239 kg/s (1.897 Mlb/hr).	16. Analyses and tests (at a test facility) will be performed.	16. Test reports and analyses exist and conclude that the DPV flow capacity is greater than or equal to 239 kg/s (1.897 Mlb/hr).
17. Vacuum breakers are provided on SRV discharge lines to reduce the post-discharge reflood height of water.	17. An inspection will be performed to confirm that the vacuum breakers are installed.	17. Vacuum breakers are installed on the SRV discharge lines. An analysis exists that demonstrates that the vacuum breaker capacity and setpoint limit the water column in the discharge line.
18. The MSIVs close upon any of the following conditions: (a) Low RPV water level, (b) Low turbine inlet pressure (RUN mode) and (c) Low main condenser vacuum.	18. Valve closure tests will be performed on the MSIVs using simulated signals.	18. The MSIVs close upon generation of any of the following simulated signals: (a) Low RPV water level, (b) Low turbine inlet pressure (RUN mode) and (c) Low main condenser vacuum.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. The ADS has an Automatic Inhibit of the automatic ADS initiation.	19. A test of the ADS will be conducted with a simulated APRM ATWS permissive signal present.	19. ADS actuation does not occur.
20. The ADS has a Manual Inhibit of the automatic ADS initiation.	20. A test of the ADS will be conducted with a generated signal of the ADS Manual Inhibit set to inhibit.	20. ADS actuation does not occur.
~~20.~~21. The ADS can be initiated manually.	~~20.~~21. Tests will be conducted by initiating each ADS division manually.	~~20.~~21. Upon receipt of a manual initiation signal, an ADS actuation signal is generated to the associated ADS valve solenoids.
~~20.~~22. The RPV water level instrumentation considers the effects of dissolved non-condensable gasses in the RPV water instrument lines.	~~20.~~22. Analyses of the as-built RPV water level instrumentation will be performed using available test data and/or operating experience.	~~20.~~22. An analysis output exists which concludes that the RPV water level instrumentation considers the effects of dissolved non-condensable gasses in the RPV water level instrument lines.
~~20.~~23. The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building is physically separated from the other divisions.	~~20.~~23. Inspections of the as-built NBS instrumentation will be conducted.	~~20.~~23. The mechanical portion of each NBS instrumentation division is physically separated from the other divisions by structural and/or fire barriers.
~~20.~~24. Motor operated valves designated in Section 2.1.2 as having an active safety function will close under differential pressure, fluid flow and temperature conditions.	~~20.~~24. Tests of installed valves for closing will be conducted under preoperational differential pressure, fluid flow and temperature conditions.	~~20.~~24. Upon receipt of an actuating signal, each motor operated valve closes.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
~~20.~~25. Control valves designated in Section 2.1.2 as having an active safety-related function will open, close or both open and close under system pressure, fluid flow and temperature conditions	~~20.~~25. Tests of the installed valves for opening, closing or both opening and closing, will be conducted under system preoperational pressure, fluid flow and temperature conditions.	~~20.~~25. Based on the direction of the differential pressure across the valve, each control valve opens, closes or both opens and closes depending upon the valve's safety function.
~~20.~~26. The ADS accumulator can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator.	~~20.~~26. An analysis and/or type test will be performed to demonstrate the capacity of the SRV ADS accumulators.	~~20.~~26. The SRV ADS accumulators have the capacity to lift the stem of the SRVs to the full open position one time with the drywell pressure at the drywell design pressure.