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Subject: **Response to NRC Request for Additional Information Letter No. 14 Related to ESBWR Design Certification Application – Reactor Vessel – RAI Numbers 5.3-1 through 5.3-12**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter. This completes GE's response to RAI Letter Number 14.

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

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

D068

Enclosure:

1. MFN 06-128 - Response to NRC Request for Additional Information Letter No. 14 Related to ESBWR Design Certification Application – Reactor Vessel – RAI Numbers 5.3-1 through 5.3-12

Reference:

1. MFN 06-093, Letter from U. S. Nuclear Regulatory Commission to Mr. David H. Hinds, *Request for Additional Information Letter No. 14 Related to ESBWR Design Certification Application,* March 22, 2006

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)
	eDRF	0000-0054-8916

Enclosure 1

MFN 06-128

Response to NRC Request for Additional Information

Letter No. 14 Related to ESBWR Design

Certification Application – Reactor Vessel

RAI Numbers 5.3-1 through 5.3-12

NRC RAI 5.3-1

The applicant stated in DCD Table 5.2-4 that both non-L and L grade types 304 and 316 stainless steel materials will be used for instrumentation nozzles and drain nozzles. Previous experience indicates that 304/316 (non-L grade) stainless steel materials are prone to intergranular stress corrosion cracking (IGSCC) when exposed to boiling water reactor (BWR) reactor coolant system (RCS) water. Therefore, staff requests the applicant justify the use of non-L grade stainless steel materials for reactor vessel (RV) components that are potentially exposed to the BWR RCS water.

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 5.2-4 between 304 and 304L/316 and 316L is only strength. A footnote will be added to DCD Table 5.2-4 indicating that for these components the maximum carbon content is 0.02%. A marked-up copy of Table 5.2-4 is shown in the following:

Table 5.2-4

Reactor Coolant Pressure Boundary Materials

• Component	• Form	• Material	• Specification (ASTM/ASME)
Main Steam Isolation Valves (MSIVs)			
Valve Body	Cast	Carbon steel	SA352 LCB
Cover	Forged	Carbon Steel	SA350LF2
Poppet	Forged	Carbon Steel	SA350LF2
Valve stem	Rod	Precipitation-hardened steel	SA 564 Gr 630 (H1100)
Body bolt	Bolting	Alloy steel	SA 540 B23 CL5
Hex nuts	Bolting Nuts	Alloy steel	SA 194 GR7
Safety/-Relief and Depressurization Valves			
Body (SRV)	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2 ASME SA 352 LCB
Body (DPV)	**Forging** [1]or Casting	Stainless Steel	SA 182, F304L or F316L **SA-351 CF 3**
Bonnet (yoke)	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2 ASME SA 352 LCB
Nozzle (seat)	Forging or Casting	**Stainless steel**[1] Carbon steel	ASME SA 182 Gr F316 ASME SA 350 LF2
Body to bonnet stud	Bar/rod	Alloy steel	ASME SA 193 Gr B7
Body to bonnet nut	Bar/rod	Alloy steel	ASME SA 194 Gr 7
Disk	Forging or Casting	Nickel alloy Stainless steel	ASME SA 637 Gr 718 ASME SA 351 CF 3A

Table 5.2-4

Reactor Coolant Pressure Boundary Materials

• **Component**	• **Form**	• **Material**	• **Specification (ASTM/ASME)**
Spring washer and Adjusting Screw or Setpoint adjustment assembly	Forging Forgings	Carbon steel Alloy steel Carbon and alloy steel parts	ASME SA 105 ASME SA 193 Gr B6 Multiple specifications
Spindle (stem)	Bar	Precipitation-hardened steel	ASTM A564 Gr 630 (H1100)
Spring	Wire or Bellville washers	Steel Alloy steel	ASTM A304 Gr 4161 N 45 Cr Mo V67
Main Steam Piping			
Pipe	Seamless	Carbon steel	SA 333 Gr. 6
Contour nozzle	Forging	Low alloy steel	SA 508 Class 3
200 mm 1500 lb. large groove flange	Forging	Carbon steel	SA 350 LF 2
50 mm special nozzle	Forging	Carbon steel	SA 350 LF 2
Elbow	Seamless	Carbon steel	SA 420
Head fitting/penetration piping	Forging	Carbon steel	SA 350 LF 2

CRD			
Middle flange	Forging	**Stainless steel** [1]	SA-182, Type or SA-336, Class F304/F304L/F316/F316L
Spool piece	Forging	**Stainless steel** [1]	SA-182, Type or SA-336 Class F304/F304L/F316/F316L

Mounting bolts	Bolting	Alloy steel	SA-193, Grade B7
Reactor Pressure Vessel			
Shells and Heads	Plate	Mn-1/2 Mo-1/2 Ni	SA-533, Type B, Class 1
	Forging	3/4 Ni-1/2 Mo-Cr-V Low alloy steel	SA-508, Grade 3, Class 1
Shell and Head Flange	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508, Grade 3, Class 1
Nozzles	Forging	3/4Ni-1/2Mo-Cr-V Low alloy steel	SA-508, Grade 3, Class 1
Drain Nozzles	Forging	Cr-Ni-Mo **Stainless steel** [1]	SA-182, Type or SA-336, Class F304/F304L/F316/F316L
Instrumentation Nozzles	Forging	Cr-Ni-Mo **Stainless steel** [1] and Ni-Cr-Fe	SA-182, Type or SA-336, Class F304/F304L/F316/F316L and Code Case N-580-1
Stub Tubes	Bar, Smls. Pipes Forging	Ni-Cr-Fe	Code Case N-580-1
Isolation Condenser			
Steam pipe	Seamless	Carbon steel	SA-333, Grade 6
Condensate pipe	Seamless	**Stainless steel** [1]	**SA-312**,Type 316L
Feedwater Piping			
Pipe	Seamless	Low Alloy	SA-335, Grade P22
Fittings	Forging	Low Alloy	SA-336, Grade F22

[1] ***The maximum carbon content is limited to 0.02%***

NRC RAI 5.3-2

The applicant stated that it will use machine/manual welding processes for bottom head penetrations. If the manual welding process, i.e. shielded metal arc welding (SMAW) is used, either 182 or 152 electrode can be used. Previous experience indicates that 182 welds are prone to IGSCC when they are exposed to the BWR RCS water. Therefore, the staff requests that the applicant provide more details in regards to the selection of appropriate welding electrodes for the fabrication of bottom head penetrations so that these welds are less prone to IGSCC. If the bottom head penetrations are to be fabricated with machine welding process provide information regarding the selection of welding wire.

GE Response

The current practice for welding stub tubes to the bottom head is automatic GTAW. The inclusion of manual welding in the DCD is to allow for local repair using manual GTAW or GMAW. All weld metal is Alloy 82 with stabilization parameter control. Use of Alloy 182 is prohibited in contact with reactor water.

Statements to clarify that use of Alloy 182 is prohibited will be added to the DCD Chapters 4.5.2.5 and 5.3.3.2.1 as shown in the following mark-ups:

4.5.2.5 Other Materials

Hardenable martensitic stainless steel and precipitation hardening stainless steels are not used in the reactor internals.

Materials, other than Type-300 stainless steel, employed in reactor internals are:

- Type or Grade XM-19 stainless steel
- Niobium modified Alloy 600 per ASME Code Case No. N-580-1
- N07750 (Alloy X-750) or equivalent

All Niobium modified Alloy 600 material is used in the solution annealed condition.

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.

Hard chromium plating surface is applied to austenitic stainless steel couplings.

All materials used for reactor internals shall be selected for their compatibility with the reactor coolant as shown in ASME Code Section III, NG-2160 and NG-3120. The fabrication and cleaning controls preclude contamination of nickel-based alloys by chloride ions, fluoride ions, sulfur, or lead.

All materials referenced in this subsection have been successfully used for many years in BWR applications.

Use of Alloy 182 is prohibited in contact with reactor water

5.3.3.2.1 Summary Description

Reactor Vessel

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

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

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

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

The welding material used for cladding in the shell area is ASME SFA 5.9 or SFA 5.4, type 309L or 309MoL for the first layer, and type 308L or 309L/MoL for subsequent layers. For the bottom head cladding, the welding material is ASME SFA 5.14, type ERNiCr-3. ***Use of Alloy 182 for welding of the CRD stub tubes in the bottom head is prohibited.***

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

The vessel head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 55.6°C (100°F) in any one-hour period. To detect seal failure, a vent tap is located between the two seal rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal ring seal.

NRC RAI 5.3-3

Per DCD Tier 2, Table 5.3-3, the nominal height from the inside of the bottom head (elevation zero) to the inside of the top head is 27.56 m (90.4 ft). The maximum corresponding height of a current BWR vessel is 21.9 m (71.8 ft). It is the staff's understanding that the vessel will be assembled at the site. Please provide a discussion of the following:

- *Final vessel assembly at the plant site is a unique concept which is quite unprecedented. Provide a detailed discussion of the planned process.*
- *If a localized heat treatment of the vessel circumferential welds are used for stress relieving, please discuss the analytical techniques that may be used to address the effect of stress distribution around the heat treated areas.*

GE Response

- Several BWR reactor vessels have been site assembled. This includes Vermont Yankee, Monticello, Leibstadt, Clinton and Limerick. The process for ESBWR has not been finalized at this time, but it is anticipated the nearly completed reactor vessel will be shipped to the site in two, or possibly three sections. Joining of the sections at the site may be done with the vessel axis vertical using mechanized welding equipment. Alternately, temporary rollers may be set up at the site and the closure weld completed with mechanized SAW or GMAW.
- Local post weld heat treatment, as allowed by ASME Section III, will be performed on the circumferential weld(s). This is a relatively simple operation because the weld joins two axisymmetric cylinders of uniform thickness. The goal is to locate the welds away from discontinuities. Finite element analysis will be used to establish the heating pattern and define temperature gradients away from the heated band. This will be followed by stress analysis to demonstrate that stresses in the adjacent material are maintained at acceptable levels. This approach has previously been successfully used to apply local PWHT to reactor vessel nozzles where reapplication of nozzle butters was required. Likewise, this approach is routinely applied to attach main steam nozzle extension forgings of low alloy steel to the steam nozzle at the ABWR construction sites.

  Local heat treatment of the final closing weld has been standard practice by some European manufacturers (e.g. Cofrentes RPV) since most of their furnaces do not have the capacity to heat treat a complete RPV. The local heat treatments were performed using either heating pads or induction heating.

NRC RAI 5.3-4

Regarding the reactor vessel (RV) surveillance capsule/holders, define the surveillance capsule lead factors and azimuthal locations in the DCD. Alternatively, add a combined operating license (COL) action item specifying that this information will be submitted in the COL application..

GE Response

As stated in DCD Section 5.3.1.6.4, the lead factor will exceed 1.0. To achieve a lead factor exceeding 1.0 is relatively easy in the ESBWR because there are no obstructions in the annulus that restrict placement of the capsule holders. The location of the axial and circumferential flux peaks are known from fluence calculations, and the capsule holders can be placed precisely at these peak locations (there are a total of eight peak locations). Since the capsule holder is mounted somewhat inboard of the vessel wall, a lead factor greater than 1.0 is assured.

An analysis defining the lead factors will be performed and the azimuth locations of the surveillance holders will be defined either as a DCD or COL action item. A description of the neutron fluence calculation is given in DCD Section 4.1.4.5.

NRC RAI 5.3-5

Because temperature affects the neutron embrittlement of the materials, provide information on the operating temperature of the vessel. If the vessel operates at a temperature below 274 °C (525 °F), discuss the effects of temperature on embrittlement of RPV materials. Please revise DCD Tier 2, Section 5.3.1.6.1 to include this information

GE Response

As with all BWRs, ESBWR will operate at a nominal temperature of ~550°F (288°C). The reactor vessel wall inside surface may be slightly lower, but not significantly, and certainly not below 525°F.

DCD Section 5.3.1.6.1 will be revised to include a statement that the RPV material will not be exposed to normal operating temperatures below 274 °C (525 °F), and that embrittlement due to temperature is of no concern as shown in the following mark-up:

5.3.1.6.1 Compliance with Reactor Vessel Material Surveillance Program Requirements

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and thermal environment. ***The RPV material will not be exposed to normal operating temperatures below 274 °C (525 °F). Therefore, embrittlement due to temperature is of no concern.***

Reactor vessel materials surveillance specimens are provided in accordance with requirements of ASTM E 185 and 10 CFR 50 Appendix H. Materials for the program are selected to represent materials used in the reactor beltline region. Specimens are manufactured from a forging actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld HAZ material. The base metal and weld are heat treated in a manner, which simulates the actual heat treatment performed on the beltline region of the completed vessel. Each in-reactor surveillance capsule contains 36 Charpy V-notch and 6 tensile specimens. The capsule loading consists of 12 Charpy V Specimens each of base metal, weld metal, HAZ material, and three tensile specimens each from base metal and weld metal. A set of out-of-reactor beltline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors are located within the capsules as required by ASTM E 185.

Four capsules are provided to consider the 60 year design life of the vessel. This exceeds the three capsules specified in ASTM E 185 as required by 10 CFR 50, Appendix H, since the predicted transition temperature shift is less than 55.6°C (100°F) at the inside of the vessel.

The following proposed withdrawal schedule is modified from the ASTM E 185 schedule to consider the 60 year design life:

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

Fracture toughness testing of irradiated capsule specimens are in accordance with requirements of ASTM E 185 as required by 10 CFR 50 Appendix H.

NRC RAI 5.3-6

Provide information regarding the derivation of the pressure-temperature (P/T) limit curves provided in the DCD. Identify any deviations from the recommended calculational procedures in Revision 1, July 1981, of Section 5.3.2 of the Standard Review Plan (SRP) and confirm conformance with Appendix G to 10 CFR Part 50.

Clarify the intent of the P/T curves provided in the DCD (e.g., representative or bounding).

GE Response

The pressure-temperature calculation is performed in accordance with the requirements of 10CFR50 Appendix G. For the representative curves provided, the material initial RT_{NDT} data from the RPV specification is used. To calculate the adjusted reference temperature (accounting for the effects of irradiation in the vessel beltline region), the copper and nickel specification limits were used in combination with the peak fluence values and the methodology of RG 1.99, Revision 2. This is considered conservative since the actual RT_{NDT} values and chemical composition are normally much lower than the ones specified. Margins for the adjusted reference temperature calculation are as defined in RG 1.99 Revision 2.

For each individual component (e.g., main steam nozzle), a finite element model is used to determine the stresses (pressure and thermal) for the transient events for normal and upset conditions. These stresses are then used to determine the applied K_I for each transient. The most limiting transient K_I for a given pressure and temperature is then compared to the minimum required K_I (note that the minimum temperature limits of 10CFR 50 Appendix G also apply). The minimum required K_I is based upon the limiting RT_{NDT} of the materials for the component (determined per above), and calculated using the methodology of ASME Section III, Appendix G. For the pressure test condition, a safety factor of 1.5 is applied to K_{Ip} (K_I from primary membrane and bending stresses). For the core not critical and core critical conditions, a factor of 2.0 is applied to K_{IP}). These safety factors are consistent with ASME Section III, Appendix G.

The P/T limit curves provided in the DCD are representative of typical BWR P/T curves.

Section 5.3.2.1 of the DCD will be revised as follows:

5.3.2.1 Limit Curves

The pressure/temperature limit curves in Figures 5.3-1 and 5.3-2 ***are representative for the ESBWR. They*** are based on the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99.

The vessel flange, RPV head and flange areas, feedwater nozzles, bottom head and the core beltline areas were ***evaluated using the material initial RT_{NDT} data from the RPV specification.***. The operating limit curves are based on the most limiting locations. The pressure/temperature limits are based on flaw sizes specified in Paragraph G-2120 of ASME Section III, Appendix G. The maximum through wall temperature gradient from continuous heating or cooling at 55.6°C (100°F) per hour was considered. The safety factors applied were as specified in ASME Section III, Appendix G.

To calculate the adjusted reference temperature (accounting for the effects of irradiation in the vessel beltline region), the copper and nickel specification limits were used in combination with the peak fluence values and the methodology of RG 1.99, Revision 2. This is considered conservative since the actual RT_{NDT} values and chemical composition are normally much lower than the ones specified. Margins for the adjusted reference temperature calculation are as defined in RG 1.99 Revision 2.

For each individual component (e.g., main steam nozzle), a finite element model is used to determine the stresses (pressure and thermal) for the transient events for normal and upset conditions. These stresses are then used to determine the applied K_I for each transient. The most limiting transient K_I for a given pressure and temperature is then compared to the minimum required K_I (note that the minimum temperature limits of 10CFR 50 Appendix G also apply). The minimum required K_I is based upon the limiting RT_{NDT} of the materials for the component (determined per above), and calculated using the methodology of ASME Section III, Appendix G. For the pressure test condition, a safety factor of 1.5 is applied to K_{Ip} (K_I from primary membrane and bending stresses). For the core not critical and core critical conditions, a factor of 2.0 is applied to K_{IP}). The RT_{NDT} of the vessel materials are determined in accordance with the ASME Section III, Subsection NB-2320, and the requirements are listed in Table 5.3-1.

Temperature Limits for Boltup

Minimum flange and fastener temperatures of RT_{NDT} plus 33°C (60°F) are required for tensioning at preload condition and during detensioning. AS shown in Table 5.3-1, this is higher than that calculated in accordance with the methods described in ASME Section III, Appendix G.

<u>**Temperature Limits for ISI Hydrostatic and Leak Pressure Tests**</u>

Representative Pressure versus temperature limits for pre-service and in-service tests when the core is not critical are shown in Figure 5.3-1. ***<u>Pressure/temperature curves using plant specific data such as materials, fluences</u> and stresses will be developed prior to plant hydrostatic test.***

Operating Limits During Heatup, Cooldown, and Core Operation

Figure 5.3-2 specifies ***representative*** limits applicable for normal reactor operation, including anticipated operational occurrences. ***Pressure/temperature curves using plant specific data such as materials, fluences and stresses will be developed prior to plant start-up.***

Reactor Vessel Annealing

In-place annealing of the reactor vessel, because of radiation embrittlement, is not necessary because the vessel is predicted to maintain an equivalent safety margin in accordance with the procedures of 10 CFR 50 Appendix G, Paragraph IVA.

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

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

The calculations are based on the limits specified in Table 5.3-1 on copper and nickel in the weld and forging material.

The fluence analysis was performed using the NRC accepted methodology documented in Reference 5.3-1. The estimated peak fluence for the vessel base material (1/4 T) and the weld above the TAF (at the inside of the RPV) are provided in Table 5.3-4.

As required by 10 CFR 50 Appendix H, a surveillance program will be conducted in accordance with the requirements of ASTM E-185. The surveillance program will include samples of base metal, weld metal and HAZ material of the beltline forging. Subsection 5.3.1.6 provides additional detail on the surveillance program.

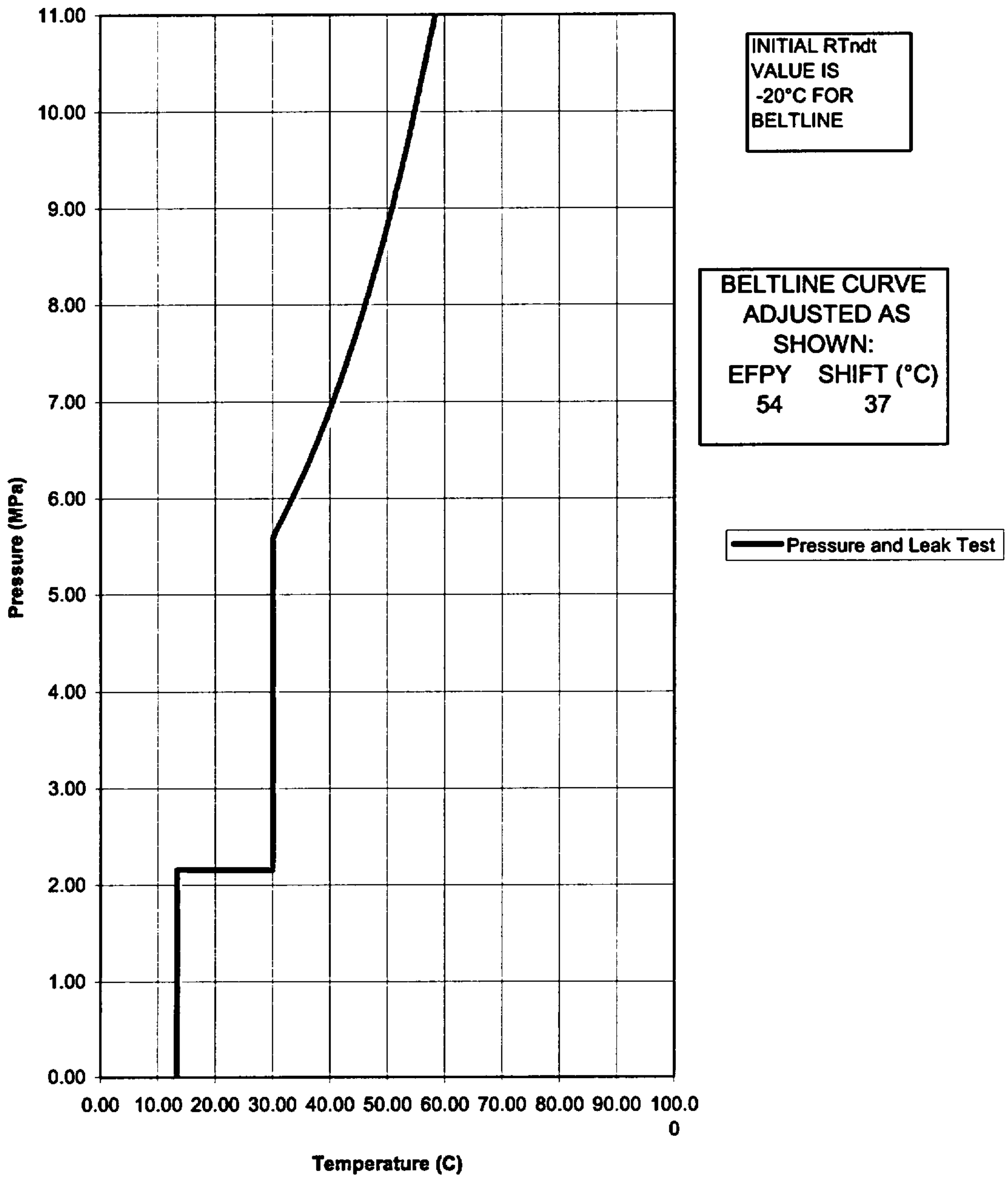

Figure 5.3-1. Minimum Temperatures Required Versus Reactor Pressure for Hydrotest — Core Not Critical ***(Representative curve for the ESBWR)***

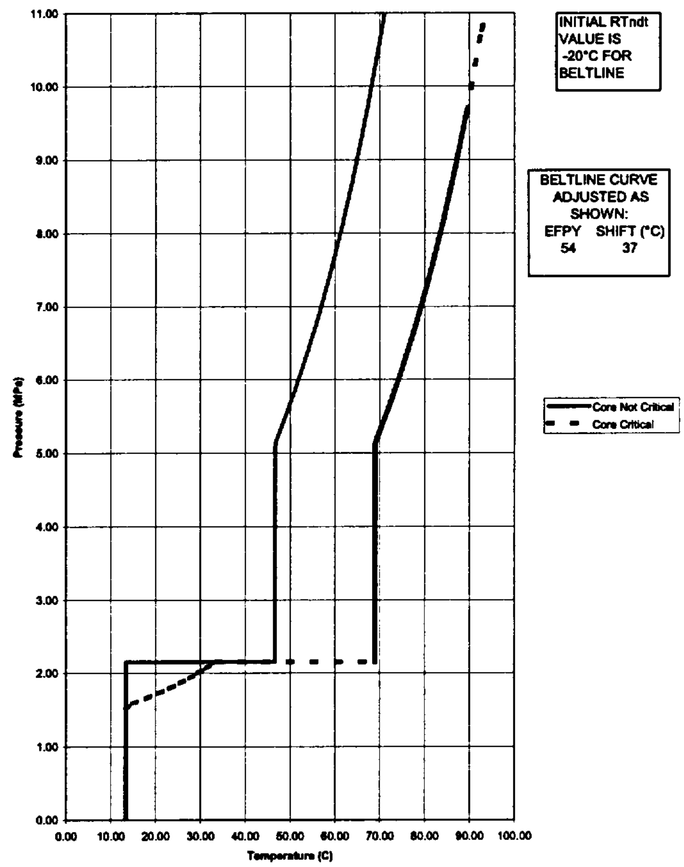

Figure 5.3-2. Minimum Temperatures Required Versus Reactor Pressure for Normal Startup and Shutdown ***(Representative curve for the ESBWR)***

NRC RAI 5.3-7

The applicant stated that the P/T limit curves for the RV will be provided by the COL applicant. The staff requests that the applicant commit, in the ESBWR DCD, that P/T limits will be generated using plant-specific data (materials, fluence etc). Also, please clarify if the P/T limit curves will be submitted by the COL applicant or holder.

GE Response

A commitment will be provided as either a COL action or as an ITAAC. This commitment will indicate the need to provide plant specific information for the development of the P/T limit curves. A statement that plant specific data are to be used will be included in the DCD as stated in the revised Subsection 5.3.4 below:

5.3.4 COL Information

Fracture Toughness Data

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

Materials and Surveillance Capsule

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

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

NRC RAI 5.3-8

The DCD indicates that the results of the material surveillance program will be used for the development of P/T limit curves. Please verify that the material surveillance program data that will be used for recalculating these curves is the plant-specific/integrated surveillance capsule program data obtained by the COL.

GE Response

The actual RPV material properties are used to refine the P-T curves prior to plant startup. The data from the surveillance capsules occurs after plant startup in accordance with the schedule defined in subsection 5.3.1.6.1 of the DCD. 10 CFR 50 Appendix H, which the COL Holder is required to follow, defines the process that is to be followed if it is necessary to change the P-T curves based on the results of the surveillance program.

NRC RAI 5.3-9

DCD Section, Tier 2, Section 5.3.1.6.5, "Time and number of Dosimetry Measurements" states that "once the fluence to thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output". Please address the following:

(a) This statement violates the provisions of Appendix H to 10 CFR Part 50.

(b) The statement is true only for equilibrium operation that would exclude the first two cycles and other cycles following fuel vendor changes or future power uprates. Does Figure 4.3-1 indicate that ESBWR power distribution in the outer assemblies may not allow future power uprates?

(c) If archival material is to be irradiated (per Appendix H to 10 CFR Part 50) is there not going to be associated dosimetry?

(d) Please identify the dosimeter compliment you propose to use in the ESBWR. Will all dosimeters have an unobstructed view of the core and provide a full spectrum coverage?

GE Response

a) Appendix H of 10CFR50 requires that an adequate dosimetry be implemented for each reactor. There are four removable surveillance specimen capsules located in surveillance capsule brackets on the ESBWR reactor vessel wall at the peak fluence locations of the core beltline region. The capsules will be withdrawn in accordance with a scheduled program over the 60-year reactor vessel design life. Each surveillance specimen capsule contains four (4) neutron dosimeter casings; one located at the top, one at the bottom and two in the middle of the capsules. Each surveillance specimen capsule also contains two temperature monitors located in the middle of the capsule. The neutron dosimeters and temperature monitors located inside the surveillance capsules meet the requirements for dosimetry of Appendix H of 10 CFR Part50 and ASTM-E185.

   The separate neutron dosimeter referenced in DCD Tier 2, Section 5.3.1.6.5 is a single removable neutron dosimeter that is located in the reactor to verify the predicted fluence-to-thermal output. This dosimeter is withdrawn at an early date in plant operation. Additional dosimeters can be inserted at a later date if necessary.

b) Figure 4.3-1 provides the basis for the current estimate of ESBWR vessel fluences from the results of neutron transport calculations. If a significant change in the core power/flow, loading pattern, or fuel type occurs in the future – e.g., power uprate or fuel vendor change – new neutron transport calculations will need to be performed to confirm/revise the fluence values.

c) Yes, if archive material is to be irradiated, there will be additional dosimetry.

d) Each dosimeter casing contains one (1) copper wire, one (1) iron wire and one (1) niobium wire, covering the spectrum of interest. There will be an unobstructed view of the core, unlike BWR/3-6 which have jet pumps that potentially obstruct the view of the core.

NRC RAI 5.3-10

DCD Section 5.3.3 refers to RG 1.2 regarding reactor pressure vessel thermal shock. However, RG 1.2 was withdrawn in 1991. Please refer another guide.

GE Response

The DCD will be revised to delete the reference to RG 1.2 as shown in revised subsection 5.3.3. There is no other guide replacing this reference.

5.3.3 Reactor Vessel Integrity

In accordance with SRP 5.3.3, Draft Revision 2, the portions of the DCD listed below are all related to the integrity of the reactor vessel. Although most of these areas are developed separately in other DCD subsections, the integrity of the reactor vessel is of such importance that a special summary discussion of all factors relating to the integrity of the reactor vessel is warranted. The information in each area is discussed to ensure that the information is complete, and that no inconsistencies in information or requirements exist that would reduce the certainty of vessel integrity.

1. Design

The basic design of the reactor vessel concerning compatibility of design with established quality standards for material properties and fabrication methods is described in Subsection 5.3.1, "Reactor Vessel Materials," establishes compatibility with required inspections as described in Subsection Subsection 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing."

2. Materials of Construction

The materials of construction are each taken into consideration as described in Subsection 5.2.3, "Reactor Coolant Pressure Boundary Materials," and in Subsection 5.3.1, "Reactor Vessel Materials."

3. Fabrication Methods

The processes used to fabricate the reactor vessel, including forming, welding, cladding, and machining, are described in Subsection 5.3.1.

4. Inspection Requirements

The inspection test methods and requirements are described in Subsection 5.3.1.

5. Shipment and Installation

Protective measures taken during shipment of the reactor vessel and its installation at the site verify that the as-built characteristics of the reactor vessel are not degraded by improper handling.

6. Operating Conditions

All the operating conditions as they relate to the integrity of the reactor vessel are described in Subsection 5.3.2, "Pressure-Temperature Limits."

7. Inservice Surveillance

Plans and provisions for inservice surveillance of the reactor vessel are described in Subsections 5.3.1 and 5.2.4.

The basic acceptance criteria for each review area are covered by other subsections, so they are discussed here only in general terms. References are made to the subsections that include detailed criteria. The acceptance criteria in these subsections describe methods that meet the requirements of the following Commission regulations in 10 CFR Part 50:

General Design Criteria 1, 4, 14, 30, 31, and 32 of Appendix A; Appendix B; 10 CFR 50.60 and associated Appendices G, and H; and 10 CFR 50.55a.

The specific criteria which meet the relevant requirements are as presented in the following subsections.

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

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

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

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

Paragraph referring to RG 1.2 deleted

The investigation shall include:

- a comprehensive thermal analysis considering the effect of blowdown and the Gravity-Driven Cooling System reflooding;
- a stress analysis considering the effects of pressure, temperature, seismic load, jetload, dead weight, and residual stresses;

- the radiation effect on material toughness (RT_{NDT} shift and critical stress intensity); and
- methods for calculating crack tip stress intensity associated with a nonuniform stress field following the design basis accident.

Appendix G of the ASME Code, Section III shall be applied as a mandatory procedure for demonstrating protection against nonductile failure. The criteria of 10 CFR 50 Appendix G are interpreted as establishing the requirements of annealing. Paragraph IVB requires the vessels to be designed for annealing of the beltline only where the existence of an adequate safety margin cannot be demonstrated in accordance with Paragraph IVA of 10CFR50 Appendix G. The ESBWR vessel is predicted to maintain an adequate safety margin throughout the life of the vessel; therefore, design for annealing is not required.

For further discussion of fracture toughness of the RPV, refer to Subsections 5.3.1.5 and 5.3.2.

NRC RAI 5.3-11

DCD Section 5.3.3.1 regarding reactor vessel failure probability should read 1 x 10 e-6 per reactor year.

GE Response

This is a typo. The DCD will be revised to specify "1.0 x 10^{-6} per reactor year" as shown in the following mark-up:

5.3.3.1 Design Bases

Safety Design Basis

The reactor vessel and appurtenances are required to withstand different combinations of loadings for loading conditions specified in the design document resulting from operation under normal and abnormal conditions.

To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

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

Power Generation Design Bases

The power generation design bases of the reactor vessel are:

- develop a simplified system that provides all safety-related functions [i.e., that failure to provide a safety function is incredible (probability of failure is less than ***1 x 10^{-6} per reactor year***)];
- develop the ESBWR vessel with a design life of 60 years with a total plant availability of 92% or greater; and
- design the reactor vessel and appurtenances which allows for a suitable program of inspection and surveillance.

NRC RAI 5.3-12

In DCD Table 5.3-4, please add the peak inside surface fluence and its corresponding azimuthal location. Please state how the 1/4T values were derived, i.e. using the RG 1.99 formula or by direct calculation?

GE Response

The DCD table will be revised to include the peak inside surface fluence and its corresponding azimuth location. as shown in the marked-up Table 5.3-4 below:

The fluences at 1/4T were calculated from direct calculations.

Table 5.3-4

RPV Fluence[12] Analysis Results

• **Parameter**	• **Value For 60 Yrs** (n/cm^2)
Expected peak neutron fluence at the 1/4 T location	• $< 1.37 \times 10^{19}$
Estimated ¼ T fluence for the weld above the TAF	• $< 4.14 \times 10^{17}$
Expected peak neutron fluence at the inside surface (n/cm^2)	$<2.07 \times 10^{19}$
Expected peak azimuthal locations (first quadrant)	***11.5°, 78.5°***

[12] ***Fluences values obtained from direct calculations.***