Enclosure 2

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NEDO-33243-A Revision 2, "ESBWR Marathon Control Rod Mechanical Design Report"

Non-Proprietary Information

HITACHI

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LICENSING TOPICAL REPORT

ESBWR MARATHON CONTROL ROD MECHANICAL DESIGN REPORT

NON-PROPRIETARY INFORMATION NOTICE

This is a public version of NEDC-33244P-A, Revision 2, from which the proprietary information has been removed. Portions of the document that have been removed are indicated by white space within double square brackets, as shown here [[ ]].

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ACRONYMS AND ABBREVIATIONS

Acronym / Abbreviation	Description
AOO	Anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
CRB	Control rod blade
DCD	Design Control Document
FMCRD	Fine Motion Control rod drive
CRDA	Control Rod Drop Accident
ECCS	Emergency core cooling system(s)
ECP	Engineering Computer Code
ESF	Engineered Safety Feature
FHA	Fuel Handling Accident
GEH	General Electric -Hitachi Nuclear Energy
GNF	Global Nuclear Fuels
IASCC	Irradiation Assisted Stress Corrosion Cracking
LOCA	Loss of Coolant Accident
LTR	Licensing topical report
MCPR	Minimum Critical Power Ratio
MSLBA	Main Steamline Break Accident
NRC	U.S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
QA	Quality assurance
RAI	Request for additional information
SRSS	Square root sum of squares
SSE	Safe Shutdown Earthquake
STS	Standard Technical Specifications
TS	Technical Specifications

EXECUTIVE SUMMARY

The GEH ESBWR Marathon control rod is a derivative of the BWR/2-6 Marathon design approved by Reference 1. The primary difference between the ESBWR Marathon and the BWR/2-6 Marathon design is a shorter absorber section appropriate for ESBWR application. The ESBWR Marathon design uses the same square absorber tube design, as the BWR/2-6 Marathon design, approved in Reference 1.

The ESBWR Marathon control rod uses a boron carbide (B_4C) capsule with the same cross-sectional dimensions as the Marathon-5S control rod (Reference 3). Compared to the original BWR/2-6 design (Reference 1), the ESBWR capsule has [[

]]

The structure of the ESBWR Marathon control rod has been evaluated during all normal and upset conditions, and has been found to be mechanically acceptable. The fatigue usage of the control rod has also been found to be well below lifetime limits.

[[

]] For all cases, the mechanical lifetime exceeds the nuclear lifetime. Therefore, the ESBWR Marathon control rod is nuclear lifetime limited.

The licensing acceptance criteria contained in the ESBWR Design Control Document (Reference 4) are evaluated and are judged to be sufficient and complete. The nuclear analysis of the ESBWR Marathon control rod is described in Reference 5. GEH requests NRC approval for the use of the Marathon control rod for ESBWR.

1. INTRODUCTION AND BACKGROUND

This report contains ESBWR Marathon control rod mechanical analysis results.

Revision 1 of this report represented a complete revision of the NEDO-33244 revision 0 report, incorporating changes to the control rod design.

NEDO-33244-A Revision 2 is the NRC-accepted version of this Licensing Topical Report. This revision includes the following:

- Changes to Section 5, Surveillance, in response to RAI 4.4-26 S01 (MFN 08-757).
- Changes to Section 4.4, Reactivity, to explain how GEH addresses control rod depletion (MFN 10-038). Corresponding changes were also made to Section 6, References.
- Addition of "-A" to the document number denoting NRC acceptance of this revision for ESBWR design certification.
- Addition of the NRC letter describing the acceptance of this revision of this Licensing Topical Report as well as Enclosure 1 of the letter, which contains the Final Safety Evaluation for this Licensing Topical Report. These items have been added at the end of this report as Attachment 1.

GEH currently manufactures the long life Marathon Control Rod Blade (CRB) for BWR/2 through BWR/6. The Nuclear Regulatory Commission (NRC) acceptance of the Marathon CRB is documented by a Licensing Topical Report (LTR), Reference 1. The Marathon CRB consists of 'square' absorber tubes, edge welded together to form the control rod wings, and welded to individual tie rod segments to form the cruciform assembly shape. The square absorber tubes are filled with a combination of boron carbide (B_4C) capsules, empty capsules, hafnium rods, and spacers. Previously, GEH manufactured original equipment and replacement Duralife Control Rod Blades, which consisted of a full-length tie rod, with boron carbide absorber rods and hafnium plates and/or strips enclosed within a sheath to form each wing. The most recent Duralife Licensing Topical Report is shown as Reference 2.

The design presented in this report is the Marathon control rod, adapted for use in ESBWR. The following sections contain the mechanical analysis of the Marathon control rod for application to ESBWR. The nuclear analysis is contained in Reference 5.

2. DESIGN CHANGE DESCRIPTION

The basic design of the ESBWR Marathon is the same as the BWR/2-6 Marathon approved by Reference 1. The control rod wings consist of edge welded square absorber tubes (Figure 2-1). The ESBWR Marathon control rod is an all boron carbide design as all tubes are filled with either boron carbide capsules, or empty capsule plenums. As in the BWR/2-6 Marathon design, the ESBWR capsules use a crimped capsule end cap connection.

There are six design changes made to the BWR/2-6 Marathon CRB, as described in Reference 1, to produce the ESBWR Marathon CRB. These changes are described in the following subsections.

2.1 ABSORBER SECTION LENGTH

Since the active fuel height of the ESBWR design is shorter than BWR/2-6, the active absorber zone for the control rod is also shorter. As shown in Table 2-1 of Reference 1, the nominal length of the BWR/2-6 Marathon absorber section is [[ ]]. For the ESBWR version, this is reduced to [[ ]]. This value is reflected in Table 2-1.

2.2 CAPSULE GEOMETRY

The ESBWR Marathon CRB uses a capsule body tube geometry with [[ ]]. The cross-sectional dimensions of the ESBWR capsule are identical to the capsule for the Marathon-5S design described in Reference 3.

A comparison of the ESBWR and the BWR/2-6 Marathon capsule dimensions is contained in Table 2-1. Due to irradiation induced B_4C powder swelling, a B_4C capsule expands as the absorber is depleted. [[

]]. This is discussed in more detail in Section 3.6

2.3 CAPSULE LENGTH

The BWR/2-6 Marathon CRB LTR (Reference 1) identifies the nominal length of the B_4C capsules as 11.4 inches. Current BWR/2-6 Marathon CRB designs use 36" capsules [[ ]] and 24" [[ ]] B_4C capsules. [[

]]

Due to the reduced length of the ESWR absorber section, the length of the capsules is also reduced. The ESBWR design uses nominal length [[ ]] boron carbide capsules. These capsule lengths are reflected in Table 2-1. A diagram of the absorber section load pattern is shown in Figure 2-2.

2.4 CONNECTOR

To be compatible with the ESBWR Fine Motion Control Rod Drive (FMCRD), the ESBWR Marathon control rod uses a connector, rather than a velocity limiter, as shown in Figures 2-3 and 2-4.

2.5 HANDLE WITH SPACER PADS

The Marathon LTR (Reference 1) allows for the use of the traditional handle with rollers or handles with wear pads. To eliminate the possibility of stress corrosion cracking initiating within the handle pin-hole, the ESBWR Marathon control rod employs a raised spacer pad, similar to what is currently being used for D lattice (BWR/2-4) Marathon control rod applications. The raised spacer pad is shown in Figure 2-4.

2.6 FULL LENGTH TIE ROD

The BWR/2-6 Marathon CRB uses multiple tie rod segments along the center of the cruciform shape. The ESBWR Marathon CRB utilizes a single tie rod that runs the entire length of the assembly similar to that used on Duralife control rods (see Reference 2). The cross-sectional geometry of this full-length tie rod is designed such that it does not alter the interface between the control rod and the adjacent fuel channels. This is achieved by ensuring that contact occurs between the wing of the control rod and the face of the fuel channel and not at the fuel channel corner and tie rod.

A cross-section of the ESBWR Marathon control rod is shown in Figure 2-1.

Table 2-1
Comparison of Typical Parameters of Marathon and ESBWR Marathon CRBs

Parameter		BWR/6 Marathon CRB [1]	ESBWR Marathon CRB
Control Rod Weight (lb)		[[	
Absorber Tubes per Wing			
Nominal Wing Thickness (in)			
Absorber Tube			
	Length (in)		
	Inside Diameter (in)		
	Nominal Thin Section Wall Thickness (in)		]]
	Material	304S	304S
	Cross-sectional area (in^2)	[[	]]
B_4C Absorber Capsule			
	Length (in)	[[	
	Inside Diameter (in)		
	Wall Thickness (in)		
	Material		
	B4C Density (g/cc)		
	B4C Density (% theoretical)		]]

1. Values from Table 2-1 of the Marathon LTR (Reference 1), except for absorber tube cross-sectional area from design calculations. Current Marathon absorber capsule lengths are also updated, see Section 2.3.

2. [[

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[[

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Figure 2-1. ESBWR Marathon CRB Cross-Section

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Figure 2-2. ESBWR Marathon Control Rod Load Pattern

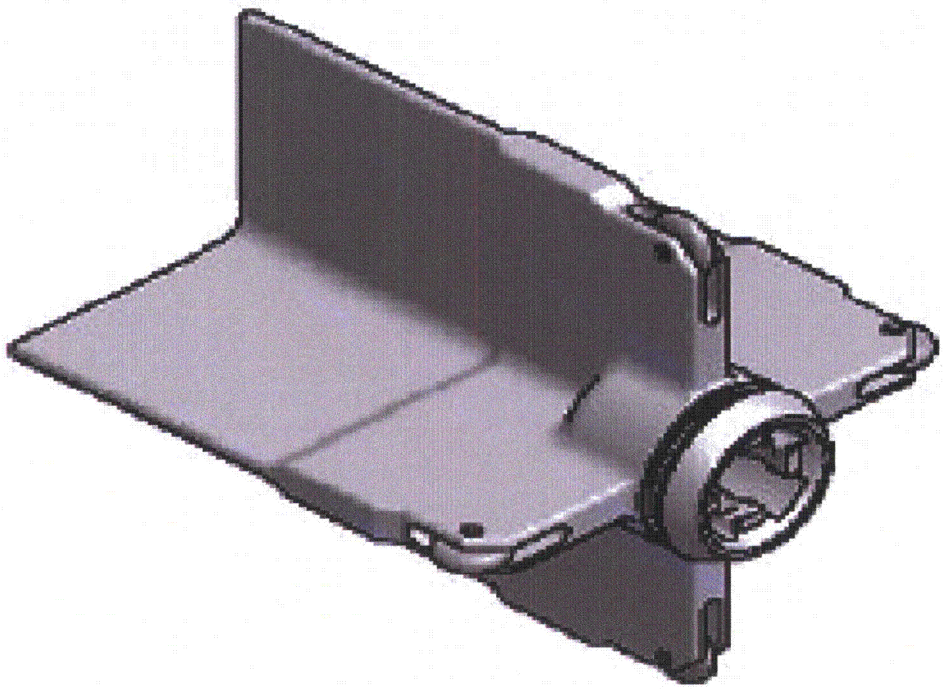

Figure 2-3. ESBWR Marathon Connector

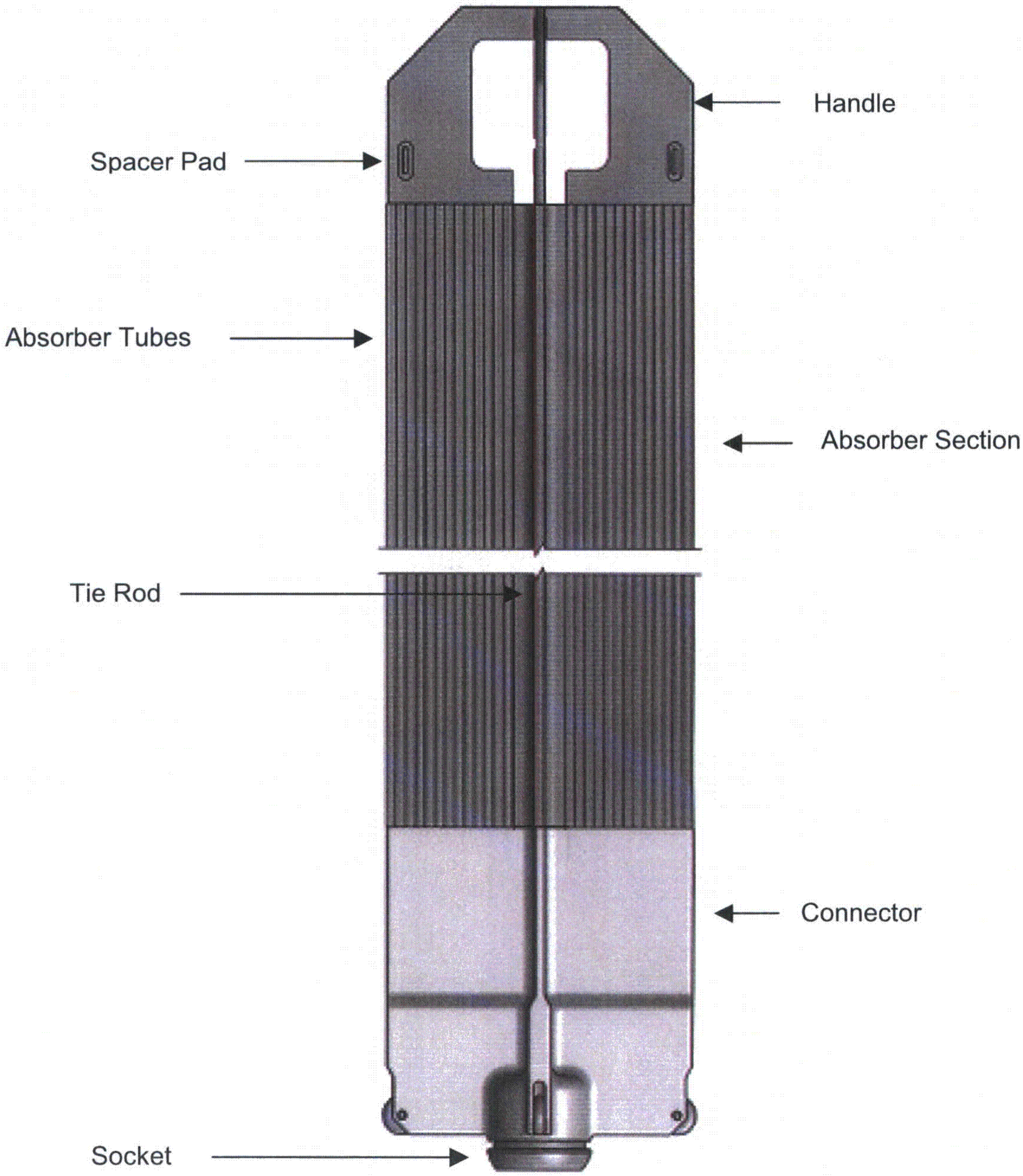

Figure 2-4. ESBWR Marathon Control Rod

3. SYSTEM DESIGN

3.1 ANALYSIS METHOD

For each control rod load application, worst case or bounding loads are identified. Stresses are calculated using worst-case dimensions and limiting material properties. For analyses involving many tolerances, square root sum of squares (SRSS) or statistical tolerancing may be used.

3.1.1 Combined Loading

As in Reference 1, effective stresses and strains are determined using the distortion energy theory (Von Mises), and compared to allowable limits. Using the principal stresses: σ_1, σ_2, and σ_3, the equivalent Von Mises stress is calculated as:

$$\sigma_{VM} = \sqrt{1/2[(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2]}$$

Both the Von Mises and Tresca stress criteria are used to predict the conditions for yielding under both uniaxial and multiaxial stress states. The Tresca Criterion can be called the maximum shear criterion since it measures the maximum shear stress present The Von Mises takes into account all principal stresses in the calculation of the conditions where yielding occurs. For thin walled tubes, under combined loads, the Von Mises Criterion appears to more accurately represent the condition under which yielding occurs (Reference 11). The use of the Von Mises criterion takes into consideration the hydrostatic component of stress and the corresponding strain value. It should be recognized that failure modes in thin walled structures such as control rod absorber tubes are initiated at the surface, a location where one of the three principal stresses is zero. The use of the von Mises criterion is therefore adequate to evaluate the potential for any of the important failure modes. First, ductile failure is associated with plastic flow. The criterion was developed to best assess that mode. Fatigue and crack growth processes would initiate on the surface. Again, plastic flow at the surface is necessary for these processes to start. As supported by the stress analyses results in Section 3.3 through 3.8, the stresses are below the un-irradiated stress limits. Therefore, the absorber tubes will only experience elastic deformation. This condition is also true in the irradiated condition where the stress ratio will decrease when compared to the actual irradiated yield strength value.

Given this, the effects of irradiation are well known. Specifically, the material will have a significant increase in yield strength and ultimate strength. Therefore, the design criteria used, one based on un-irradiated properties, will insure that as fluence is accumulated, the component continues to remain elastic and well below the actual yield strength. As stated in Reference 1, this approach has been previously accepted.

3.1.2 Unirradiated Versus Irradiated Material Properties

Each structural analysis is first evaluated to determine whether unirradiated or irradiated material properties are appropriate. In general, as stainless steel is irradiated, the yield and ultimate tensile strengths increase, while the ductility, or allowable strain decreases. In order to determine the correct technique, the analyses are divided into two categories:

1. Analyses with an applied load (i.e., scram). For these analyses, a maximum stress is calculated, and compared to the limiting unirradiated stress limit.
2. Analyses with an applied displacement (i.e., seismic bending). For these analyses, a maximum strain is calculated, and compared to the limiting irradiated strain limit.

Austenitic stainless steels do not display a ductile to brittle transition (DBTT). The material fracture toughness and ductility (in the unirradiated condition) does not vary significantly in the temperature range of interest (70 - 550°F). In turn, the effect of irradiation on austenitic stainless steel is to reduce the toughness and ductility somewhat; however, austenitic stainless steel still retains ductility after irradiation. There are existing data at high fluence that confirm the tensile ductility and fracture toughness. Specifically, ductility levels and fracture toughness data for irradiated components are documented in Reference 9. These data substantiate their ductile behavior at both room temperature as well as operating temperature.

3.2 MATERIAL PROPERTY LIMITS

The limiting unirradiated material strengths are first identified for the control rod structural materials, and shown in Table 3-1. For most materials, limiting values from the ASME Boiler and Pressure Vessel Code are used. In other cases, minimum material strengths are specified in GEH material specifications.

GEH requires that the mechanical properties of all material used in the fabrication of control rods be certified as meeting material specification limits. For example, the mechanical properties of finished, annealed, and un-irradiated type 304S absorber tubes are defined by a fabrication specification. These mechanical limits, along with the certification results of three recent absorber tube lots are shown in Table 3-22. As shown, all mechanical properties meet the specification requirements. See section 3.2.4 for more information on GEH's stabilized type 304S stainless steel.

3.2.1 Stress Criteria

The licensing acceptance criteria of Appendix 4C of Reference 4 are used, in which the control rod stresses and strains and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material, structure, or welded connection.

The figure of merit employed for the stress-strain limit is the design ratio, where:

Design ratio = effective stress/stress limit, or, effective strain/strain limit.

The design ratio must be less than or equal to 1.0. Conservatism is included in the evaluation by limiting stresses for all primary loads to one-half of the ultimate tensile value.

Resulting allowable stresses for primary loads are shown in Table 3-2.

3.2.2 Absorber Tube Material Isotropy

The irradiation resistant special melt austenitic stainless steel (type 304S) used for the control rod absorber tubes is manufactured using standard industrial processes and solution annealing. There is no significant anisotropy produced in wrought product by these procedures. Photos of finished absorber tubes, at 300X magnification, in different orientations, are shown in Figures 3-14 through 3-16. The axial loading direction is the direction of design concern and is aligned with the direction of standard tensile tests on irradiated material. The necking observed in these irradiated tensile tests can be interpreted as supporting the adequacy of the strength and ductility of the material in the radial direction.

3.2.3 Welded Connections

For welded connections, a weld quality factor, q, is used to further reduce the allowable stress. Therefore, the allowable stress for a welded connection, Sm', is:

$$S_m' = (q)Sm$$

Weld quality factors are determined based on the inspection type and frequency of the weld. Weld quality factors are shown in Table 3-3.

3.2.4 Laser Welding Process

Laser Beam Weld (LBW) processes are used extensively in the manufacture of Marathon control rods. Welding processes for control rods are developed and qualified against a set of acceptance standards which includes: (1) meeting minimum penetration requirements, (2) smooth blends between welded members, and (3) no cracks, holes, lack of fusion or porosity. Since the ESBWR Marathon CRB uses the same square absorber tube as the BWR/2-6 Marathon CRB, the weld processes are the same.

As a result of the complexity of the control rod geometry, GEH qualifies the welding process in a manner meeting the intent of the ASME Code. The qualification method selected is to confirm the mechanical properties of the weld by using a representative mockup of the laser weld. Mechanical tests confirm that the mechanical properties of the weld were higher than the minimum properties of the base metal.

The weld quality factor (q) provides a safety margin against manufacturing defects during processing. The critical to quality components of the weld are defined by ASME B&PV code weld procedure QW-264.1, Welding Procedure Specifications, Laser Beam Welding (LBW). GEH further refines its internal critical to quality requirements from the ASME B&PV code for its day-to-day operations. [[

]]

GEH performs metallographic evaluation on sample laser welds on a weekly basis to confirm that the results of the welding process remain within parameters. These results are documented. Photomicrographs of a typical laser weld, taken as part of a recent qualification test, are shown in Figure 3-16. Comparing the grain structure at the edge of the weld to an area away from the weld shows that there is no effective heat affected zone for a laser weld. This combined lack of heat affected zone, Ta stabilization, and low carbon chemistry, accounts for the good carbide test results mentioned above.

Austenitic stainless steels have no inherent age hardening capability and lend themselves readily to the welding process. GEH's proprietary Type 304 S composition is as follows:

[[

]]

A common concern in austenitic stainless steel welds is carbide precipitation. Carbide formation in a weld heat affected zone would encourage intergranular stress corrosion cracking in this location. The combination of low heat input welding practices, tantalum stabilization, and restrictive carbon limits, provides an effective barrier to such intergranular cracking.

3.2.5 Absorber Tube Axial Shrink Due to Welding

Due to the absorber tube-to-tube laser welding process, the absorber tubes shrink by varying amounts in the axial direction. The resulting residual strain is evaluated using data from production BWR/2-6 Marathon control rods. Prior to welding, the length of the BWR/2-6 absorber tube is [[ ]]. The lengths of the absorber tubes after welding were measured on a production Marathon control, and are recorded in Table 3-23.

As shown in the table, the biggest difference in relative length between the absorber tubes after welding is [[ ]]

The length of the finished BWR/2-6 absorber section is [[ ]]. Therefore, the maximum axial strain due to the differential weld shrinking of the absorber tubes is:

Strain (ε)= $\Delta L/L_{initial}$ = [[ ]]

A [[ ]]strain is metallurgically insignificant in terms of driving microstructural changes in the bulk tubing. This strain is an elastic driver towards overall distortion. Distortion is minimized through production controls. Please see section 3.2.4 for further discussion with regard to the mechanical properties of the laser welds.

3.3 SCRAM

The largest axial structural loads on a control rod blade are experienced during a control rod scram, due to the high terminal velocity. To be conservative, structural analyses of the control rod are performed assuming a 100% failed control rod drive buffer. A dynamic model of mass, spring and gap elements is used to simulate a detailed representation of the load bearing components of the assembly during a scram event. Simulations are run at atmospheric temperatures, pressures, speeds, and properties as well at operating temperatures, pressures, speeds, and properties. The resulting loads are shown in Table 3-4.

Structural stresses are determined from the scram loads shown in Table 3-4 using the limiting material properties, weld quality factors, and worst-case geometry for the area subject to the load. Figure 3-1 shows the welds and cross-sections analyzed.

Resulting maximum stresses during a failed buffer scram are shown in Table 3-5. These stresses are evaluated against the stress limits shown in Table 3-2. Specific details for each calculation are shown in Appendix A. As shown by the design ratios in Table 3-5, sufficient margin exists against failure for all cross-sections and welds.

3.4 SEISMIC AND FUEL CHANNEL BOW INDUCED BENDING

Fuel channel deflections, which result from seismic events, impose lateral loads on the control rods. The ESBWR Marathon control rod is analyzed for the most limiting Safe Shutdown Earthquake (SSE) event.

3.4.1 Wing Outer Edge Bending

The SSE analysis is performed by evaluating the strain in the ESBWR Marathon absorber section with maximum SSE deflection. In addition, maximum control rod deflections due to fuel channel bulge and bow are conservatively added to the calculated seismic bending deflections. [[

]].

The limiting location for strain due to bending of the control rod cross-section occurs at the outer edge of the control rod wing. At this location, a combined strain due to simultaneous application of the following loads is calculated: (1) control rod bending due to an SSE seismic event, (2) control rod bending due to worst case channel bulge and bow, (3) axial absorber tube stress due to maximum internal pressure, and (4) a failed buffer scram. The results of these strain calculations are shown in Table 3-6. As shown, even under these combined worst-case conditions, the maximum strain is well below the limiting maximum allowable strain at irradiated conditions.

3.4.2 Absorber Tube to Tie Rod Weld

The combined effect of control rod bending due to SSE and channel bulge and bow deflection combined with maximum absorber tube internal pressure is also evaluated at the full-length tie rod to absorber tube weld. A finite element model is used, as shown in Figure 3-2. Resulting worst-case stresses are shown in Table 3-7. As shown, the resulting stresses are acceptable against the design criteria.

3.4.3 Absorber Tube Lateral Load

Finally, the lateral load imposed on the control rod absorber tube due to an excessively bowed channel is evaluated. The finite element model is shown in Figure 3-11. As shown, the entire lateral load is applied to the wear surface of a single square absorber tube, along with reactor internal pressure. For conservatism, no internal pressure is applied to the tube, which would offset the external pressure and reduce the stresses in the tube.

The resulting stress intensity plot is shown in Figure 3-12. The maximum stress intensity is calculated as [[ ]], which is less than the absorber tube allowable load of [[ ]] from Table 3-2.

The lateral load model is also evaluated using end-of-life irradiated material properties. This analysis is extremely conservative, since for the tube to be irradiated, there would be a corresponding build-up of internal pressure in the tube to offset the lateral load. However, for this model, no internal pressure is applied. The results of the calculation is a maximum stress intensity [[ ]], which is less than ½ of the irradiated true ultimate strength of the material of [[ ]]. Further, the maximum strain intensity is [[ ]], compared to an ultimate strain of [[ ]].

3.5 STUCK ROD COMPRESSION

Maximum compression loads from the Fine Motion Control Rod Drive (FMCRD) are evaluated for a stuck control rod. Both buckling, and compressive yield are analyzed for the entire control rod cross-section (buckling mode A), and conservatively assuming that the entire compression load is applied to a single control rod wing (buckling mode B). Figure 3-3 shows the buckling modes.

Results of the stuck rod compression loads are contained in Table 3-8 for the entire control rod cross-section (mode A), and in Table 3-9 for the single wing (mode B). As can be seen, neither compressive yielding nor buckling will occur for either buckling mode. Additionally, for both buckling modes, the compressive yield load is reached prior to the critical buckling load.

3.6 ABSORBER BURN-UP RELATED LOADS

The structure of a control rod must provide for positioning and containment of the neutron absorber material (Boron Carbide powder, Hafnium, etc) throughout its nuclear and mechanical life and prohibit migration of the absorber out of its containment during normal, abnormal, emergency and faulted conditions. The ESBWR Marathon CRB contains boron carbide powder within capsules contained within absorber tubes (capsule within a tube design).

The boron neutron absorption reaction releases helium atoms. Some of this helium gas is retained within the compacted boron carbide powder matrix, causing the powder column to swell. This swelling causes the B_4C capsule to expand. The remainder of the helium is released as a gas. The capsule end caps for the ESBWR Marathon design are crimped to the capsule body tubes. This allows the helium gas to escape from the capsule and fill the absorber tube gap and any empty capsule plenum volume provided.

For the BWR/2-6 Marathon capsule design, [[

]].

For the ESBWR capsule design, [[

]].

Using the pressurization capability of the absorber tube, limits are determined for each absorber tube configuration (see Figure 2-2), in terms of B_4C column depletion.

These individual absorber tube depletion limits are then combined with radial depletion profiles and axial depletion profiles to determine the mechanical depletion limit for the control rod assembly. See Section 3.9.

3.6.1 Irradiated Boron Carbide Swelling Design Basis

Mechanical test data of the irradiated behavior of boron carbide was obtained by irradiating test capsules for a period of approximately ten years in a reactor. Test capsules were placed in neutron monitor tubes and irradiated in a reactor. The configurations of two types of test capsules used are shown in Figure 3-7.

The dimensions of the test capsules were measured prior to irradiation, and post-irradiation in a hot cell using standard laboratory practice. For test capsules with a mandrel, the diametral strains were mathematically corrected to compensate for the mandrel, resulting in an increase of reported strain value.

Diametral swelling results are shown in the Table 3-15 and Figure 3-8. The ESBWR Marathon swelling analysis conservatively uses the $+3\sigma$ upper bound value of [[ ]].

Axial swelling data is shown in Table 3-16. As shown, the axial swelling is [[

]].

3.6.2 Clearance Between Capsule and Absorber Tube

As a result of the welding process forming the control rod wings, the inside diameter of the absorber tubes shrink. Therefore, a minimum inside diameter is established, and is 100% inspected following the welding, before the absorber section is loaded with capsules.

To evaluate the clearance between the capsule and absorber tube, worst-case capsule dimensions are used, which result in the maximum outside diameter at 100% local depletion. These consist of the original maximum outside diameter, and minimum wall thickness, resulting in the maximum beginning boron carbide diameter

The strain at the ID of the capsule is equal to the diametral strain of the boron carbide powder. The $+3\sigma$ upper limit of [[ ]] from Table 3-15 is used. Then, assuming constant volume deformation of the capsule, the strain on the outside diameter of the capsule is:

[[ ]].

Then, the capsule outside diameter at 100% local depletion is:

$OD_{100\%} = OD_0(1+ \varepsilon_{OD})$.

A summary of this calculation is shown in Table 3-17. As shown, at 100% local depletion, using worst-case capsule and absorber tube dimensions and a conservative boron carbide swelling basis, a clearance exists between the capsule and the absorber tube. Therefore, there is no strain placed on the outer absorber tube due to boron carbide swelling.

3.6.3 Thermal Analysis

Pressure in the absorber tube due to helium release is calculated accounting for worst-case capsule and absorber tube dimensions and B_4C helium release fraction. Because the fraction of helium released from the B_4C powder increases with temperature, a finite element thermal analysis is performed to determine the peak B_4C temperature (see Figure 3-5). This thermal analysis is performed using worst-case dimensions, maximum end-of-life crud buildup, combined with maximum beginning-of-life heat generation.

For the thermal model, corrosion is modeled as the build-up of an insulating layer of crud. This crud may be corrosion products from the control rod absorber tube, or deposited from other reactor internals. For all thermal analyses, a [[ ]] thick crud layer is applied, which is twice that assumed for the BWR/2-6 Marathon design.

A temperature distribution is shown in Figure 3-5. The model used assumes that the tube is interior to the wing, in that there is another absorber tube to the left and right. The boundary on the left and right is conservatively assumed to be insulated (zero heat flux).

Results of the thermal analysis are shown in Table 3-10, and in Figure 3-5. The following conservatisms are applied to the thermal model:

- Peak beginning-of-life heat generation rates are used, these are combined with:
- End-of-life combined corrosion and crud build-up of [[ ]], twice that used in previous analyses.
- Peak heat generation rates are used from the highest heat generation tube, which is actually the outermost edge tube. In reality, this tube will have coolant on one side, rather than be insulated. Further some heat transfer will occur from the peak heat generation tube to the adjacent tube, rather than be perfectly insulated.
- Maximum wall thickness dimensions are used.

Peak B_4C temperatures are shown in Table 3-10. The temperatures shown in this table are based on peak beginning-of-life boron carbide heat generation rates (see Reference 5), and are from the peak heat generation absorber tube at the peak axial location. They are radially averaged only across the cross-section of an individual boron carbide capsule.

Helium release fractions are based on models developed using data from multiple sources. The data shows a significant dependence of helium release fraction on the irradiation temperature. The helium release fractions used are shown in Table 3-10. The helium release model is based on data from 500 °F to 1000 °F, which envelopes the temperatures shown in Table 3-10.

3.6.4 Absorber Tube Pressurization Capability

[[

]] Finite element analyses are performed to determine the pressurization capability of the absorber tube.

The burst pressure is defined as the internal pressure at which any point in the tube reaches a stress intensity equal to the true ultimate strength of the material. Then, to calculate an allowable pressure, a safety factor of 2.0 is applied to the differential pressure across the absorber tube wall such that:

$$P_{allow} = \frac{\left(P_{burst} - P_{external}\right)}{2} + P_{external}$$

The burst pressure capability of the tube is initially calculated using square absorber tube nominal dimensions. The resulting burst pressure is then scaled down by [[ ]] to match burst pressure testing results. The nominal dimension and scaled burst pressures are shown in Table 3-20.

The pressurization analysis is then performed at worst-case drawing dimensions. The resulting burst pressure is shown in Table 3-20. As shown, although the burst pressure is less than the nominal case, it is bounded by the scaled burst pressure used to determine the tube allowable pressure. Therefore, the design basis absorber tube allowable pressure is conservative.

The effect of the welded connection of the innermost absorber tube to the tie rod is evaluated by modifying the pressurization model to incorporate the tie rod (Figure 3-10). The resulting burst pressure for this model is shown in Table 3-20. As shown, the burst pressure is less than the nominal, single tube value. However, it is bounded by the scaled burst pressure used to determine the absorber tube allowable pressure. Therefore, the design basis absorber tube allowable pressure is conservative.

Maximum Stress Components

Stress components at the point of maximum stress intensity were analyzed for the absorber tube with the maximum allowable internal pressure. The point of maximum stress intensity is found to be on the inside surface of the absorber tube. Principle stress components are shown in Table 3-18. All stress values shown in Table 3-18 are within the allowable stress value for 304S tubing of [[ ]] shown in Table 3-2.

Effect of the Welded Connection Between Absorber Tubes

The Marathon Control Rod Blade (CRB) is manufactured using very low heat input laser weld processes. The resulting regions of microstructural change including the associated heat affected zones (HAZ) are very small (see section 3.2). Based on general understanding, the fine HAZ microstructure will have mechanical properties that are equivalent to, or exceed, those of the wrought base material. Therefore, the HAZ will have mechanical properties that exceed the required minimum properties of the associated wrought material.

Two potential issues arise from welding of the absorber section: (1) sensitization and (2) residual stress. These issues are addressed below:

Sensitization: The low heat input laser welding processes have minimal impact on the wrought tube material, in that they typically do not result in sensitized material. To confirm this conclusion, the processes are continually evaluated metallographically to confirm the acceptability of the weld region (i.e., lack of sensitization). In addition, [[

]]. Note also from section 3.6.2 that these contact hoop stresses (and associated strains) have been eliminated for the ESBWR Marathon control rod.

Residual stress: One major effect of the welding process is that it will introduce tensile residual stresses in the narrow weld/HAZ region. These stresses are not a significant concern for two reasons: (1) The field cracking has not been associated with the weld HAZ and (2) the irradiation experienced by the CRB over the initial time of operation can significantly reduce these stresses by 60% or more through radiation creep processes (Reference 10). At this level of reduced stress, there is little concern for any effect on stress corrosion cracking (SCC) initiation or their applied stresses and strains. In that the major concern are strains from swelling, this level of stress is well below those levels required to even produce yielding. See also section 3.2.

<u>Absorber Tube Expansion</u>

The pressurization of the absorber tubes will cause an axial expansion of the tubes. This is due to the internal pressure pushing against the end plugs that seal the ends of the absorber tubes. Using the maximum allowable internal pressure, the area of the end plugs, and the number of pressurized tubes in the absorber section, the maximum axial load is calculated and shown in Table 3-19.

Assuming stresses remain in the elastic range, the axial strain on the absorber tubes is calculated as $\varepsilon = \sigma/E = P/AE$, with the elongation being $\Delta L = \varepsilon L$. For an absorber section that is nominally [[ ]] long, the total elongation is also shown in Table 3-19. These maximum elongations are relatively small, and will not affect the fit, form or function of the control rod.

<u>Effect of Irradiated Material</u>

The pressurization finite element model uses unirradiated material properties. To test the assertion that the use of unirradiated properties in the pressurization finite element model is conservative, two test cases are performed using irradiated material properties: (1) the single tube model used to establish the burst pressure, and (2) the model incorporating the absorber tube to tie rod weld. For each test, the unscaled burst pressure of [[ ]] (Table 3-21) is applied. Resulting peak stress and strain intensities are determined, and are compared to material allowables in Table 3-21.

As shown in Table 3-21, the resulting stress and strain intensities are less than the material ultimate values. Since the [[ ]] burst pressure is based on the unirradiated material reaching the true ultimate stress, it may be stated that the unirradiated material analysis for absorber tube pressurization is generically conservative.

3.6.5 Irradiation Assisted Stress Corrosion Cracking Resistance

In order for the stress corrosion cracking mechanism to activate it requires a material that is susceptible, a conducive environment and a sustained tensile stress. If one of these three mechanisms is not present to a sufficient degree, the likelihood of a stress corrosion crack to form is significantly reduced.

The Marathon absorber tube is made from a GEH proprietary stainless steel, "Rad Resist 304S", which is optimized to be resistant to Irradiation Assisted Stress Corrosion Cracking (IASCC). The chemistry of this material is shown in Section 3.2.4.

In addition to using IASCC resistant material, the ESBWR Marathon control rod is designed such that the swelling of the boron carbide capsule does not impart a mechanically imposed stress/strain on the absorber tube. See section 3.6.2. This significantly reduces the amount of stress/strain present in the absorber tubes at the end of life, and significantly reduces the likelihood of stress-corrosion cracking.

3.7 HANDLING LOADS

The ESBWR Marathon control rod is designed to accommodate three times the weight of the control rod during handling, to account for dynamic loads. The handle is analyzed using a finite element model, using worst-case geometry (see Figure 3-6). Table 3-11 shows the results of the handle loads analysis.

3.8 FATIGUE

The ESBWR control rod is designed to withstand load combinations including anticipated operational occurrences (AOOs) and fatigue loads associated with those combinations. The fatigue analysis is based on the following assumed lifetime:

[[

]]

For scram, each cycle represents a single scram insertion. Scram simulations show that the oscillations in the control rod structure damp out quickly. Further, it is extremely conservative to assume [[ ]] scrams with a 100% inoperative control rod drive buffer, as the loads experienced by the control rod in a normal buffered scram are much less severe.

For the Safe Shutdown Earthquake (SSE), a total of [[ ]] seismic events is assumed, in which each event consists of [[ ]] cycles of control rod lateral bending. The assumption of [[ ]] lifetime SSE events is also considered very conservative.

Based on the reactor cycles, the combined loads are then evaluated for the cumulative effect of maximum cyclic loadings. The fatigue usage is evaluated against a limit of 1.0. The maximum cyclic stress is determined using a conservative stress concentration factor of 3.0 for welded connections. Table 3-12 shows the fatigue usage due to control rod SCRAM at six locations. In this analysis, it is assumed that each scram occurs with a 100% failed FMCRD buffer.

Table 3-13 shows the fatigue usage at the control rod outer edge due to bending from SSE seismic events and severe channel bow, control rod scram, and maximum absorber tube internal pressure. As can be seen, the combined fatigue usage is much less than 1.0.

Table 3-14 shows the fatigue usage at the tie rod to first absorber tube weld. The combined loading due to failed buffer scram, maximum absorber tube internal pressure, SSE seismic events and severe channel bow is considered. As shown, the combined fatigue usage is much less than 1.0.

It is well known that the cycles for fatigue initiation are dependent on the stress or strain range. The stress amplitudes are all in the elastic range. As shown in Tables 3-12 through 3-14, based upon the ASME Section III fatigue design curve for un-irradiated austenitic material (Reference 6), the low number of cycles represents only a small amount of cumulative damage, well below the design limit. The ½ ultimate tensile stress value represents the ASME design limit for ~30,000 cycles. It has been established that an increase in the strength level, consistent with the effect of irradiation, would only increase the margin. This is supported by data on high strength materials, which confirm that the endurance limit is close to ½ ultimate tensile stress (Reference 7).

The last consideration with regard to fatigue is an evaluation of whether there is any flow-induced vibration that could in turn provide the potential for fatigue initiation. An assessment was performed to evaluate the loads induced by transverse loading. The evaluation that treated the control blade as a cantilever beam, found that the loads were very small and would not be sufficient to even close the gap between the blade and the fuel assembly. This load is considered so small as to be negligible, and would not lead to any risk of fatigue.

3.9 CONTROL ROD MECHANICAL LIFETIME

As discussed in Section 3.6, the lifetime limiting mechanism for the ESBWR Marathon control rod is [[ ]]. An absorber tube mechanical limit as a function of average B-10 per cent depletion is calculated based on peak heat generation, temperatures and helium release fractions, combined with worst-case component geometries. As discussed in Section 3.6, the method for evaluating the swelling phenomenon of irradiated boron carbide is very conservative, using worst-case capsule and absorber tube dimensions, along with a +3σ upper limit swelling rate assumption. Using these conservatisms, the ESBWR Marathon capsule is designed to result in a clearance between the absorber tube and capsule and 100% local depletion, thereby eliminated swelling induced strain in the outer absorber tube.

The calculation of the control rod mechanical lifetime limit, in terms of a four-segment average B-10 depletion, is shown in Table 3-23. Along the top of the table is the absorber tube number, where tube 1 is the first absorber tube, welded to the cruciform tie rod. Also shown are the span-wise radial peaking factors, which show the relative absorption rate of each absorber tube. A limiting axial depletion profile is used to calculate the B-10 depletion for each absorber tube and axial node. At the bottom of the table, the average depletion for each tube is shown, along with the depletion limit for that tube, which varies depending on the number of empty capsule plenums employed at the bottom of the absorber column. Through an iterative process, the peak ¼ segment depletion is raised until the limiting absorber tube reaches its mechanical limit. The 4-segment mechanical lifetime of the control rod is then the average of the four ¼ segments.

As shown in Table 3-23, the 4-segment mechanical lifetime limit is [[ ]]. From Reference 5, the ¼-segment nuclear depletion limit is [[ ]]. [[ ]], the nuclear lifetime of the ESBWR Marathon control rod is limiting, in that the mechanical lifetime exceeds the nuclear lifetime.

Table 3-1
ESBWR Marathon Material Properties

Material Type	Control Rod Components	Ultimate Tensile Strength, S_U (ksi)		Yield Strength, S_Y (ksi)		Modulus of Elasticity, E (x 10^6 psi)		Poisson's Ratio, ν	
		70 °F	550 °F	70 °F	550 °F	70 °F	550 °F	70 °F	550 °F
316 Plate	Handles and pads	[[							
316 Bar	Handle pads								
XM-19 Bar	Connector socket								
CF3 Casting	Connector casting								
ER 308L	Capsule end caps, absorber tube end plugs, weld filler metal								
304S Bar	Tie rods								
304S Tubing	Absorber Tubes								
Hardened 304L Tubing	Capsule body tubes								]]

Table 3-2
Design Allowable Stresses for Primary Loads

Material Type	CR Components	½ Ultimate Tensile Stress S_m (ksi)	
		70 °F	550 °F
316 Plate	Handles and pads	[[	
316 Bar	Handle pads		
XM-19 Bar	Connector socket		
CF3 Casting	Connector casting		
ER 308L	Capsule end caps, absorber tube end plugs, weld filler metal		
304S Bar	Tie rods		
304S Tubing	Absorber Tubes		
Hardened 304L Tubing	Capsule body tubes		]]

Table 3-3
Weld Quality Factors

Weld	Weld Inspection	Weld Quality Factor, q
Socket to Connector	[[	
Connector to Absorber Section		
Handle to Absorber Section		
End Plug to Absorber Tube		]]

Table 3-4
Maximum Control Rod Failed Buffer SCRAM Dynamic Loads

Components	Maximum Equivalent Loads in Kips (10^3 lbs)	
	70 °F	550 °F
Coupling	[[	
Connector		
Connector/Absorber Section Interface		
Absorber Section		
Handle/Absorber Section Interface		]]

Table 3-5
ESBWR Marathon Failed Buffer SCRAM Stresses

Location (Figure 3-1 Section)	Room Temperature (70 °F)			Operating Temperature (550 °F)		
	Maximum Stress	Allowable Limit	Design Ratio	Maximum Stress	Allowable Limit	Design Ratio
Socket Minimum Cross-Sectional Area (A-A)	[[					
Socket to Connector Weld (B-B)						
Connector Minimum Cross-Sectional Area (C-C)						
Connector to Absorber Section Weld (D-D)						
Absorber Section (E-E)						
Handle to Absorber Section Weld (F-F)						]]

Table 3-6
Outer Edge Bending Strain due to Seismic and Channel Bow Bending, Internal Absorber Tube Pressure and Failed Buffer Scram

Description	Value at 550 °F
Outer Edge Bending Strain, Seismic (%)	[[
Outer Edge Bending Strain, Seismic + Channel Bow (%)	
Max Internal Pressure Axial Stress (ksi)	
Max Failed Buffer Scram Stress (ksi)	
Total Outer Edge Strain, Seismic + Failed Buffer Scram + Absorber Tube Internal Pressure (%)	
Total Outer Edge Strain, Seismic + Channel Bow + Failed Buffer Scram + Absorber Tube Internal Pressure (%)	
Allowable Strain (%) ½ Ultimate, Irradiated	
Design Ratio	]]

Table 3-7
Absorber Tube to Tie Rod Weld Stress

Description	Value at 550 °F
Seismic + Internal Pressure, Max S_{INT} (ksi)	[[
Seismic + Channel Bow + Internal Pressure, Max S_{INT} (ksi)	
True Ultimate Tensile Stress (ksi)	
Design Ratio	]]

Table 3-8
Stuck Rod Compression Buckling – Entire Control Rod (Mode A)

Description	70 °F	550 °F
Critical Buckling Load, P_{cr} (lb)	[[	
Compressive Yield Load (lb)		
Maximum Stuck Rod Compression Load (lb)		
Design Ratio,Buckling		
Design Ratio, Compressive Yield		]]

Table 3-9
Stuck Rod Compression Buckling – Control Rod Wing (Mode B)

Description	70 °F	550 °F
Critical Buckling Load, P_{cr} (lb)	[[	
Compressive Yield Load (lb)		
Total Compressive Load (lb)		
Design Ratio, Buckling		
Design Ratio, Compressive Yield		]]

Table 3-10
Boron Carbide Peak Temperatures and Helium Release Fractions

Parameter	Nominal Dimensions	Worst Case Dimensions
B_4C Centerline Temperature (°F)	[[	
Average B_4C Temperature (°F)		
Helium Release Fraction (%)		]]

Table 3-11
Handle Lifting Load Stress

Description	Value at 70 °F
Maximum Equivalent Stress (ksi)	[[
Allowable Stress (ksi)	
Design Ratio	]]

Table 3-12
Fatigue Usage due to Failed Buffer Scram

Location	Stress Amp. (ksi)	Allowable Cycles (N)	Actual Cycles	Usage
Socket Minimum Area	[[			
Socket to Connector Weld				
Connector to Absorber Section Weld				
Absorber Section				
Handle to Absorber Section Weld				]]

Table 3-13
Fatigue Usage at Absorber Section Outer Edge

Stress Type	Stress Amp. (ksi)	Allowable Cycles (N)	Actual Cycles	Usage
Absorber Section Outer Edge - Scram + Internal Pressure	[[			
Absorber Section Outer Edge – Seismic + Channel Bow				]]
	Total Usage =		[[	]]

Table 3-14
Fatigue Usage at Absorber Tube to Tie Rod Weld

Stress Type	Stress Amp. (ksi)	Allowable Cycles (N)	Actual Cycles	Usage
Absorber Tube to Tie Rod Weld - Scram	[[			
Absorber Tube to Tie Rod Weld – Seismic + Channel Bow + Internal Pressure				]]
	Total Usage =		[[	]]

Table 3-15
Irradiated Boron Carbide Diametral Swelling Data

[[

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Table 3-16
Irradiated Boron Carbide Axial Swelling Data

[[

]]

Table 3-17
Irradiated Boron Carbide Capsule Swelling Calculation

Parameter	Value
Absorber Tube ID Before Welding (in)	[[
Minimum Absorber Tube ID After Welding (in)	
Capsule OD (in)	
Capsule Wall Thickness (in)	
Maximum Capsule OD_0 (in)	
Maximum Capsule ID_0 (in)	
Capsule ID strain (in/in)	
Capsule OD strain (in/in)	
Capsule OD at 100% local depletion	]]

Table 3-18
Absorber Tube Pressurization Results: Principle Stress Results at Operating Temperature and Pressure and Maximum Allowable Pressure

Stress Component	Value
S1 (Hoop)	[[
S2 (Axial)	
S3 (Radial)	
Equivalent Stress	
Allowable Stress	]]

Table 3-19
Control Rod Axial Elongation due to Absorber Tube Pressurization

Parameter	Value
Axial Load due to Pressurization (kips)	[[
Absorber Section Cross-Sectional Area (in^2)	
Modulus of Elasticity, E (ksi)	
Strain (in/in)	
Elongation, ΔL (inch)	]]

Table 3-20
Absorber Tube Burst and Allowable Pressures at 550 °F

Parameter	Pressure (psia)
FEA Burst Pressure, Nominal Dimensions	[[
FEA Burst Pressure, Tie Rod Model	
FEA Burst Pressure, Worst-Case Dimensions	
Scaled Burst Pressure (7% reduction to match burst pressure tests)	
Allowable Pressure (2.0 Safety Factor)	]]

Table 3-21
Absorber Tube Pressurization Using Irradiated Material

Case	Peak Stress Intensity (ksi)	True Ultimate Stress (ksi)	Peak Total Strain Intensity (%)	True Ultimate Tensile Strain (%)
Single Tube, Irradiated	[[			
Tube and Tie Rod, Irradiated				]]

Table 3-22
Type 304S Absorber Tube Mechanical Properties

Property	Room Temperature Yield Stress (ksi)	550 °F Yield Stress (ksi)	Room Temperature Ultimate Tensile Stress (ksi)	550 °F Ultimate Tensile Stress (ksi)	Room Temperature Elongation (% in 2 inches)
Specification Requirement*	[[				
Example Lot 1					
Example Lot 2					
Example Lot 3					]]

* These material requirements are specified in the fabrication specification for the absorber tubes. The tubing supplier certifies each lot of absorber tubes as meeting these requirements.

Table 3-23
Mechanical Lifetime Calculation

[[

]]

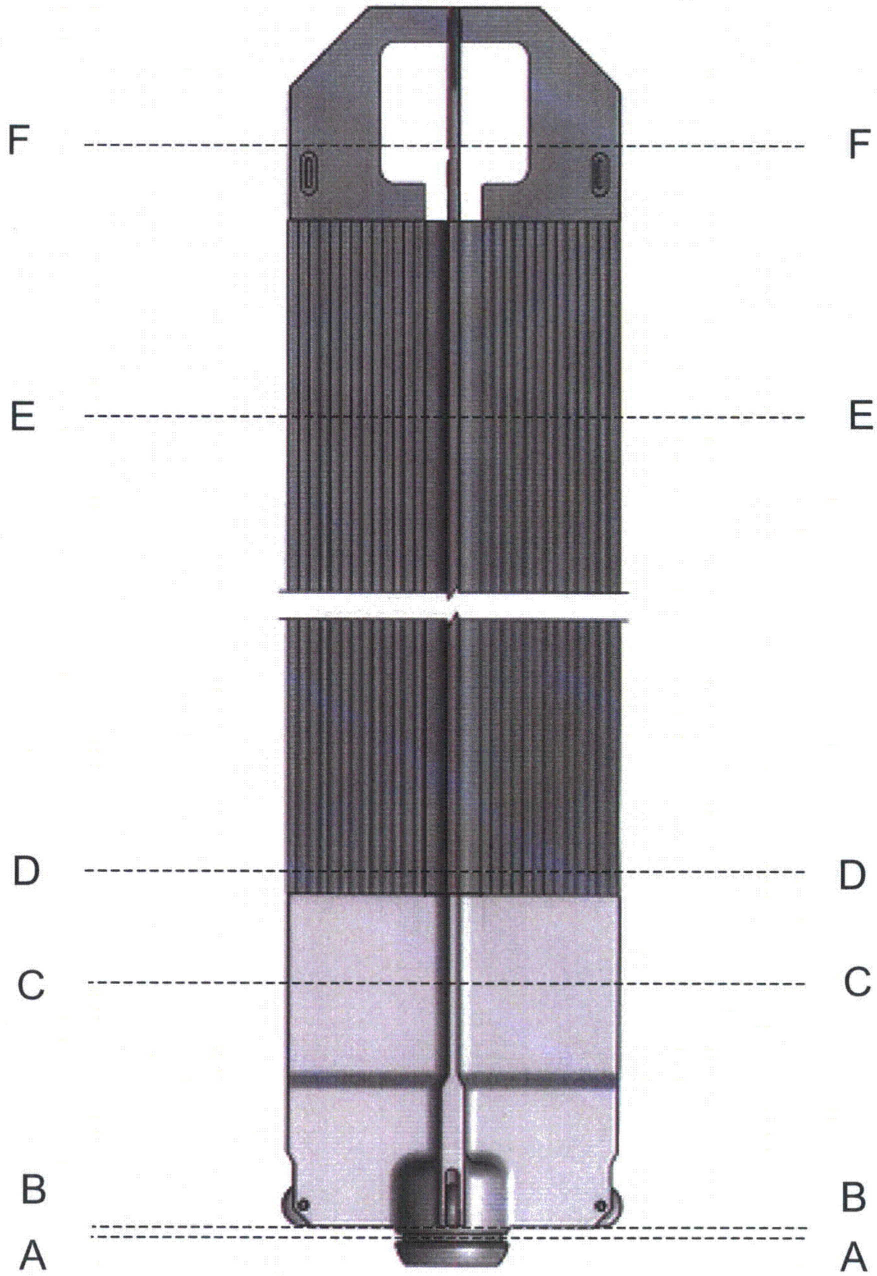

Figure 3-1. Control Rod Assembly Welds and Cross-Sections Analyzed for SCRAM

[[

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Figure 3-2. Absorber Tube to Tie Rod Finite Element Model

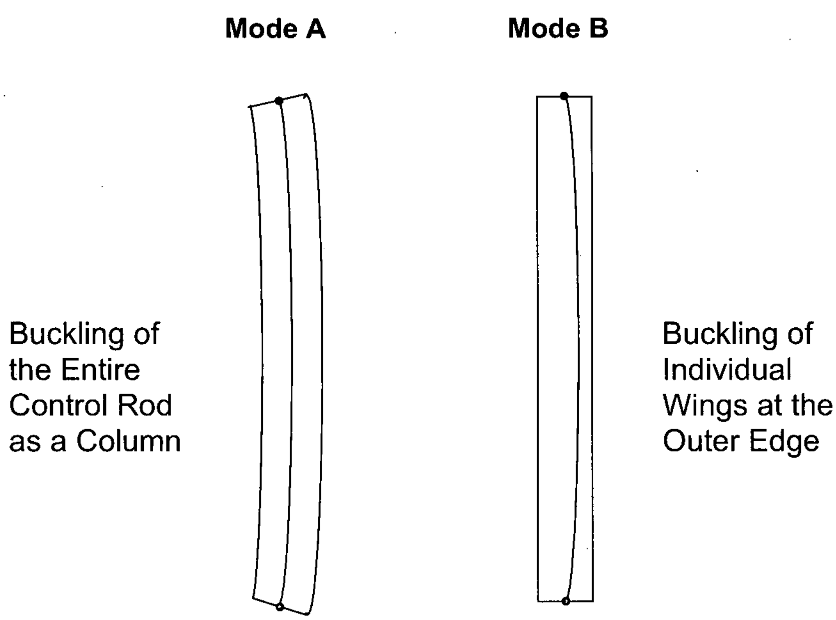

Figure 3-3. Control Rod Buckling Modes

[[

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Figure 3-4. Absorber Tube Pressurization Finite Element Model

[[

]]

Figure 3-5. Absorber Tube and Capsule Thermal Finite Element Model

[[

]]

Figure 3-6. Handle Lifting Loads Finite Element Model

[[

]]

Figure 3-7. Irradiated Test Capsule Configurations

[[

]]

Figure 3-8. Irradiated Boron Carbide Diametral Swelling Data

[[

]]

Figure 3-9. Neutron Radiograph of Irradiated Marathon Absorber Capsules

[[

]]

Figure 3-10. Tube Pressurization Finite Element Model, Tube + Tie Rou

[[

]]

Figure 3-11. Lateral Load Finite Element Model

[[

]]

Figure 3-12. Lateral Load Finite Element Results

[[

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Figure 3-13. Absorber Tube Material, 300X Magnification

[[

]]

Figure 3-14. Absorber Tube Material, 300X Magnification

[[

]]

Figure 3-15. Absorber Tube Material, 300X Magnification

[[

]]

Figure 3-16. Typical Autogenous Laser Weld of 304S Absorber Tube

4. LICENSING CRITERIA

The Design Control Document for ESBWR (Reference 4) identifies four criteria for the licensing and evaluation of the ESBWR control rod. These criteria are evaluated as follows.

4.1 STRESS, STRAIN, AND FATIGUE

4.1.1 Criteria

Control rod stresses, strains, and cumulative fatigue are evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection.

4.1.2 Conformance

As discussed in Section 3, the ESBWR Marathon design has been evaluated using the same or more conservative design bases and methodology than the Marathon CRB. All components of the control rod are found to be acceptable when analyzed for stresses due to normal, abnormal, emergency, and faulted loads. The design ratio, which is the effective stress divided by the stress limit or the effective strain divided by the strain limit, is found to be less than or equal to 1.0 for all components. Conservatism is included in the evaluation by limiting stresses for all primary loads to one-half of the ultimate strength (i.e., a safety factor of two is employed).

The fatigue usage of the ESBWR Marathon control rod is calculated using the same methodology as the Marathon CRB. The fatigue analysis assumes [[ ]]. It is found that the calculated fatigue usage is less than the material fatigue capability (the fatigue usage factor is much less than 1.0).

4.2 CONTROL ROD INSERTION

4.2.1 Criteria

The control rod design is evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.

4.2.2 Conformance

The ESBWR Marathon control rod is designed to withstand maximum stresses and strains experienced during control rod insertion, including scram. Section 3 demonstrates the structural acceptability of the ESBWR Marathon control rod.

The ability of the ESBWR Marathon control rod to insert into the core within acceptable scram times is discussed in section 4.2.4.2 of the ESBWR Tier 2 DCD (Reference 4). The worst-case scenario for a control rod scram within scram time requirements is a scram during a seismic event. As discussed in the ESBWR Tier 2 DCD (Reference 4), an ABWR Marathon control rod was tested during scram with simulated seismic fuel channel oscillation. This ABWR Marathon control rod inserted within scram time requirements, and suffered no detrimental damage. As noted in the ESBWR Tier 2 DCD (Reference 4), the ESBWR Marathon control rod seismic conditions are bounded by the ABWR test.

4.3 CONTROL ROD MATERIAL

4.3.1 Criteria

Control rod materials are shown to be compatible with the reactor environment.

4.3.2 Conformance

No new materials are introduced for the ESBWR Marathon control rod that have not been used in control rods in operating BWR/2-6 plants. The ESBWR Marathon control rod is designed to be crevice-free, and uses materials resistant to corrosion and stress corrosion cracking. For example, the absorber tubes are made from the same high purity, stabilized type 304S stainless steel as BWR/2-6 Marathon control rods. This material was developed by GEH to be resistant to stress corrosion cracking.

4.4 REACTIVITY

4.4.1 Criteria

Control rod reactivity worth shall be included in the plant core analyses.

4.4.2 Conformance

As discussed in Section 1 of Reference 5, the equilibrium core design for ESBWR was performed using a BWR/6 (S lattice) original equipment control rod. As also discussed in Reference 5, the compatibility of the ESBWR Marathon control rod is ensured by matching the initial cold reactivity worth of the Marathon CRB with the BWR/6 original equipment used in the core design.

The control rod depletion limit of 10% worth reduction in any axial quarter segment is the lifetime criterion for all approved GE/GEH control rod designs. This same limit is applied to the ESBWR Marathon control rod, as discussed in Section 2.1 of Reference 5. The 10% worth reduction limit is also documented in Section 4.2.1.1.8 of the US Supplement to GESTAR II (Reference 13).

The shutdown margin (SDM) demonstration requirement in the ESBWR Technical Specifications specifies that a demonstration be performed after fuel reconfiguration to assure that the core can remain subcritical by 0.38% $\Delta k/k$ with the strongest control rod withdrawn, as specified in Section 3.1.1 of the ESBWR Technical Specification (Reference 12). This demonstration will include any reactivity variations associated with actual control rod inventory.

GEH imposes a 1% SDM design criteria in its design and licensing process. This additional margin accommodates a number of factors that are not explicitly modeled, including the variation in control rod depletion within the allowable 10% criterion. This design margin provides assurance that sufficient SDM is present to account for the various operational and methodology uncertainties, and that the SDM demonstration when performed at the plant will have a high degree of certainty of success. Section 2.3 of Reference 14 provides a broader discussion of this.

The reactivity effects of control rod depletion on core performance during one plant operating cycle are small and are accounted for by the critical eigenvalue normalization process performed for each plant operating cycle. The cold critical eigenvalue used to calculate shutdown margin in core design and licensing analyses is determined from the most recent plant cold critical data. This plant data includes the small reactivity variations due to rod burnup for the specific control rod inventory in the reactor. Any rod replacements performed during the subsequent refueling outage will invariably increase the rod worth at those locations and provide a small decrease in cold reactivity.

The cold critical eigenvalue behavior from SDM demonstration cases is procedurally reviewed for all reactor cores with a GNF fuel supply. [[

]] GNF has never experienced a situation where the SDM demonstration failed to meet the technical specification when using GNF fuel and design & licensing methodology. The rod worth restrictions - to match the OEM rod worth within 5% at beginning of life and to limit rod worth reduction to less than 10% of the OEM worth – ensures that the impact on cold reactivity and shutdown margin remains small compared to design margins.

5. SURVEILLANCE

As directed by NRC, the following is the proposed surveillance program for the ESBWR Marathon control rod.

- The four (4) fleet-wide, highest depletion, ESBWR Marathon control rods will be tracked.
- These four (4) control rods will be visually inspected during each refueling outage, until they have achieved as close to nuclear end-of-life as practical (target minimum 90% of nuclear end-of-life).
- During refueling outages in which the depletion of the lead ESBWR Marathon assemblies are less than 75% of design nuclear life, the four (4) highest depletion ESBWR Marathon control rods will be visually inspected in-core, with two diagonal fuel bundles removed. This will allow for inspection of four of eight control rod faces, one face from each wing. Alternately, the control rods may be moved and inspected in the buffer or spent fuel pool.
- For refueling outages in which the depletion of the lead ESBWR Marathon assemblies are greater than 75% of design nuclear life, the four (4) highest depletion ESBWR Marathon control rods will be moved to the buffer or spent fuel pool, with a visual inspection of all eight faces of each control rod performed. Lead ESBWR Marathon control rods may exceed 75% depletion prior to the eight-face inspections planned in the buffer or spent fuel pool as long as those inspections are performed before the control rods are utilized in another fuel cycle.
- The in-core and fuel pool visual inspections shall have sufficient resolution, lighting, and scan rate such that crack indications similar to those observed on BWR/2-6 Marathon control rods would be seen.
- To confirm the end-of-life performance of the ESBWR Marathon control rod, the first twelve (12) control rods shall be visually inspected upon discharge, not to exceed four (4) control rods from any single plant. These visual inspections shall consist of an inspection of all eight faces of the control rod.
- Should a material integrity issue be observed, GEH will (1) arrange for additional inspections to determine a root cause and (2) if appropriate, recommend a revised lifetime limit to the NRC based on the inspections and other applicable information available.
- GEH will report to NRC the results of all ESBWR Marathon visual inspections at least annually.

6. REFERENCES

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A. APPENDIX A – FAILED BUFFER SCRAM STRESS EVALUATION

Failed buffer scram stress calculations for all cross-sections shown in Figures 3-1 and 3-2 are shown in Table 3-5. During a control rod scram, large axial loads are imparted on the control rod. These axial loads are determined using a dynamic spring and mass model, the results of which are presented in Table 3-4. For this analysis, the scram loads are determined assuming a 100% inoperative control rod drive buffer. The following cross-sections are analyzed.

A-1 SOCKET MINIMUM CROSS-SECTIONAL AREA (FIG. 3-1, SECTION A-A)

The minimum cross-sectional area of the socket is calculated from the drawing to be [[ ]]. Actual and allowable stress calculations are shown in Table A-1. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

A-2 SOCKET TO CONNECTOR WELD (FIG. 3-1, SECTION B-B)

The socket is screwed into the connector casting, and sealed using a circumferential fillet weld. The weld joins the XM-19 socket to the type CF3 connector casting, with ER 308L filler metal required. The minimum, combined effective normal area for this connection is calculated to be [[ ]]. Table A-2 calculates the actual and allowable stresses for this weld. As shown, all design ratios are less than 1.0. Therefore, the weld is acceptable.

A-3 CONNECTOR MINIMUM CROSS-SECTIONAL AREA (FIG. 3-1, SECTION C-C)

The minimum cross-sectional area of the connector is calculated from the drawing to be [[ ]]. Actual and allowable stress calculations are shown in Table A-3. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

A-4 CONNECTOR TO ABSORBER SECTION WELD (FIG. 3-1, SECTION D-D)

The weld connecting the absorber section to the connector is analyzed using the combined loading of the scram loads and axial loads due to the maximum allowable internal pressure of the absorber tubes.

Since both the scram loads and the load due to the internal pressure of the absorber tubes is considered, a combined weld area of the absorber section to connector weld, and the end plug to absorber tube weld is calculated. Since the end plug weld is in shear for this loading, the weld area is multiplied by $(1/\sqrt{3})$ to calculate an effective normal weld area. This is added to the minimum absorber section to connector weld area, which is determined using CAD software:

A_{normal} = (# of tubes){$(1/\sqrt{3})(\pi)OD_{plug,min}$(weld penetration) + (absorber section to handle/connector area per tube)}.

The weld area per tube is then multiplied by the number of tubes. The weld area calculation is summarized in Table A-4.

Once the effective normal weld area is known, the combined maximum stresses due to scram and internal pressure are calculated as described in Table A-5. As shown, all design ratios are less than 1.0. Therefore, the weld is acceptable.

A-5 ABSORBER SECTION (FIG. 3-1, SECTION E-E)

The minimum cross-sectional area of the absorber section is calculated in Table A-6. Actual and allowable stresses are shown in Table A-7. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

A-6 ABSORBER SECTION TO HANDLE WELD (FIG. 3-1, SECTION F-F)

The weld connecting the absorber section to the handle is analyzed using the combined loading of the scram loads and axial loads due to the maximum allowable internal pressure of the absorber tubes.

The effective weld area calculation is identical to the calculation in Table A-4 for the connector to absorber section weld. Using this effective normal weld area, the combined maximum stresses due to scram and internal pressure are calculated as described in Table A-8. As shown, all design ratios are less than 1.0. Therefore, the structure is acceptable.

Table A-1. Socket Axial Stress Calculations

Description	Source	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[	
Max Failed Buffer Scram Stress (ksi)	[[ ]]		
Allowable Stress (ksi)	Table 3-2 (XM-19)		
Design Ratio	=stress/allow		]]

Table A-2. Socket to Connector Weld Stress Calculations

Description	Source	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[	
Max Failed Buffer Scram Stress (ksi)	[[ ]]		
Allowable Stress (ksi)	Table 3-2 (CF3)		
Design Ratio	=stress/allow		]]

Table A-3. Minimum Connector Area Stress Calculations

Description	Source	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[	
Max Failed Buffer Scram Stress (ksi)	[[ ]]		
Allowable Stress (ksi)	Table 3-2 (CF3)		
Design Ratio	=stress/allow		]]

Table A-4. Connector to Absorber Section Weld Geometry

Description	Reference	Value
Absorber Tube to Connector Weld Area (in^2)	CAD analysis	[[
Min End Plug OD (in)	Drawing	
Max End Plug OD (in)	Drawing	
Min End Plug Weld Penetration (in)	Assembly Drawing	
Total Normal Weld Area Per Tube	CAD Analysis	
Number of Absorber Tubes per Assembly	Assembly Drawing	
Total Weld Area (in^2)	=(# tubes)(area)	]]

Table A-5. Connector to Absorber Section Weld Stress Calculations

Description	Source	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[	
Maximum Allowable Internal Pressure (ksi)	Finite Element Analysis		
End Plug Pressure Area (in^2)	$=\pi/4*(OD_{plug})^2$		
Number of Pressurized Tubes	Assembly Drawing		
Total Axial Load (kips)	=Scram Load + (press)(area)(# tubes)		
Total Weld Area (in^2)	Table A-4		
Max Failed Buffer Scram + Internal Pressure Stress (ksi)	=Ptot/A		
Allowable Stress (ksi)	Table 3-2 (CF3/304S Tubes)		
Weld Quality Factor	Table 3-3		
Allowable Weld Stress (ksi)	$=S_m*q$		
Design Ratio	=Stress/Allow		]]

Table A-6. Absorber Section Geometry Calculation

Description	Source	Value
Min Absorber Tube Area (in^2)	CAD Analysis	[[
Min Tie Rod Area (in^2)	CAD Analysis	
Number of Absorber Tubes	Assembly Drawing	
Total Minimum Absorber Section Cross-sectional Area (in^2)	=(# tubes)(tube area) + tie rod area	]]

Table A-7. Absorber Section Stress Calculation

Description	Source	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[	
Max Failed Buffer Scram Stress (ksi)	=P/A		
Allowable Stress (ksi)	Table 3-2 (304S Tubes)		
Design Ratio	=stress/allow		]]

Table A-8. Absorber Section to Handle Weld Stress Calculations

Description	Source	70 °F	550 °F
Max Failed Buffer Scram Load (kips)	Table 3-4	[[	
Maximum Allowable Internal Pressure (ksi)	Finite Element Analysis		
End Plug Pressure Area (in^2)	$=\pi/4*(OD_{plug})^2$		
Number of Pressurized Tubes	From assembly drawing		
Total Axial Load (kips)	=Scram Load + (press)(area)(# tubes)		
Total Weld Area (in^2)	Table B-10		
Max Failed Buffer Scram + Internal Pressure Stress (ksi)	=Ptot/A		
Allowable Stress (ksi)	Table 3-2 (304S Tubes)		
Weld Quality Factor	Table 3-3		
Allowable Weld Stress (ksi)	$=S_m*q$		
Design Ratio	=Stress/Allow		]]

NEDO-33244-A Revision 2

Attachment 1

NRC SAFETY EVALUATION (Non-Proprietary)

ESBWR MARATHON CONTROL ROD MECHANICAL DESIGN REPORT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 7, 2010

10-254

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE Hitachi Nuclear Energy
3901 Castle Hayne Road MC A-18
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION FOR LICENSING TOPICAL REPORTS NEDE-33243, REVISION 2, "ESBWR MARATHON CONTROL ROD NUCLEAR DESIGN" AND NEDE-33244P, REVISION 1, "ESBWR MARATHON CONTROL ROD MECHANICAL DESIGN REPORT"

Dear Mr. Head:

On August 24, 2005, GE Hitachi (GEH) Nuclear Energy submitted the Economic Simplified Boiling Water Reactor (ESBWR) design certification application to the staff of the U.S. Nuclear Regulatory Commission. Subsequently, in support of the design certification, GEH submitted the license topical reports (LTRs) NEDE-33243P, Revision 2, "ESBWR Marathon Control Rod Nuclear Design" and NEDE-33244P, Revision 1, "ESBWR Marathon Control Rod Mechanical Design Report." The staff has now completed its review of NEDE-33243P, Revision 2 and NEDE-33244P, Revision 1.

The staff finds NEDE-33243P, Revision 2, "ESBWR Marathon Control Rod Nuclear Design" and NEDE-33244P, Revision 1, "ESBWR Marathon Control Rod Mechanical Design Report," acceptable for referencing for the ESBWR design certification to the extent specified and under the limitations delineated in the LTRs and in the associated safety evaluation (SE). The SE, which is enclosed, defines the basis for acceptance of the LTR.

The staff requests that GEH publish the revised version of the LTRs listed above within 1 month of receipt of this letter. The accepted version of NEDE-33243P and NEDE-33244P shall incorporate this letter and the enclosed SE and add an "-A" (designated accepted) following the report identification number.

If NRC's criteria or regulations change, so that its conclusion that the LTR is acceptable is invalidated, GEH and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for continued applicability of the LTR without revision of the respective documentation.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosures, this document is "DECONTROLLED."

Pursuant to 10 CFR 2.390, we have determined that the enclosed SE contains proprietary information. We will delay placing the non-proprietary version of this document in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any additional information in Enclosure 1 is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390.

The Advisory Committee on Reactor Safeguards (ACRS) subcommittee, having reviewed the subject LTR and supporting documentation, agreed with the staff's recommendation for approval following the May 18, 2010 ACRS subcommittee meeting.

Sincerely,

David B. Matthews, ~~Director~~ for
Division of New ~~Reactor Licensing~~ Licensing
Office of New Reactors

Docket No. 52-010

Enclosure:
1. Safety Evaluation (Non-Proprietary)
2. Safety Evaluation (Proprietary)

cc: See next page (w/o enclosure)

cc:

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Page 3 of 3

SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS
NEDE-33244P, REVISION 1, "ESBWR MARATHON CONTROL ROD MECHANICAL DESIGN REPORT," AND NEDE-33243P, REVISION 2, "ESBWR CONTROL ROD NUCLEAR DESIGN,"
GE HITACHI NUCLEAR ENERGY, LLC

1.0 INTRODUCTION

By application dated November 15, 2007 (Reference 1), and supplemented by information in References 2, 3, and 4, GE Hitachi Nuclear Energy (GEH) requested review and approval of NEDE-33244P, Revision 1, "ESBWR Marathon Control Rod Mechanical Design Report" (Reference 5). This licensing topical report (LTR) provides mechanical analysis results for the economic simplified boiling-water reactor (ESBWR) Marathon control rod blade (CRB). This report represents a complete revision of NEDE-33244P, Revision 0 June 2006 report, and includes changes to the CRB design. Mark-ups to this topical report for inclusion in Revision 2 are also listed in Reference 1.

By application dated July 2008, GEH requested review and approval of NEDE-33243P, Revision 2, "ESBWR Control Rod Nuclear Design" (Reference 6). This LTR presents the ESBWR Marathon CRB nuclear analysis. This report represents a complete revision of the NEDE-33243P, Revision 1 June 2006, report, and includes changes to the CRB design.

The proposed ESBWR CRB design is evaluated to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures as set forth in Title 10 CFR Part 100.

2.0 REGULATORY EVALUATION

Regulatory framework for the review of fuel system designs and reactivity control systems are General Design Criteria (GDC) 10, 26, 27, and 35 in Appendix A, "General Design Criteria for Nuclear Power Plants," to Reference 18. GDC 10, "Reactor Design," establishes specified acceptable fuel design limits (SAFDLs) that should not be exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). GDC 26, "Reactivity Control System Redundancy and Capability," requires two independent reactivity control systems of different design principles including control rods capable of reliably controlling reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded. GDC 27, "Combined Reactivity Control Systems Capability," and GDC 35, "Emergency Core Cooling," establish requirements for combined reactivity control system capability and emergency core cooling capability under postulated accident conditions.

Regulatory guidance for the review of fuel system design and adherence to the GDC listed above appears in NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design," issued March 2007 (Reference 7). In accordance with SRP Section 4.2, the objectives of fuel system safety review are to ensure the following:

Enclosure 1

- The fuel system is not damaged as a result of normal operation and AOOs.
- Fuel system damage is never so severe as to prevent control rod insertion when it is required.
- The number of fuel rod failures is not underestimated for postulated accidents.
- Coolability is always maintained.

GEH proposed a set of acceptance criteria for evaluating CRB designs in 26A6642A, Revision 5, "ESBWR Design Control Document," Tier 2, Chapter 4, "Reactor," issued May 2008 (Reference 9):

- Control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection.
- The control rod design shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.
- Control rod materials shall be shown to be compatible with the reactor environment.
- Control rod reactivity worth shall be included in the plant core analyses.

3.0 TECHNICAL EVALUATION

The following summarizes the objectives of the review by the staff of the U.S. Nuclear Regulatory Commission (NRC) of NEDE-33243P, Revision 2, and NEDE-33244P, Revision 1:

- Provide assurance that the CRB design criteria are consistent with the regulatory criteria identified in SRP Section 4.2.
- Verify that the CRB design complies with the licensing acceptance criteria proposed by GEH in Reference 1, Section 4, and Reference 9, Section 4C.
- Verify that the CRB design satisfies regulatory requirements.
- Verify that the mechanical design methodology is capable of accurately or conservatively evaluating each component with respect to the applicable design criteria.
- Verify that the GEH analysis supports the mechanical lifetime reported in the LTR. If necessary, implement a surveillance program to monitor in-reactor behavior and confirm the design calculations.
- Verify that the structure of the CRB is evaluated during all normal and upset conditions and is found to be mechanically acceptable.
- Verify that the methods and models used to calculate nuclear lifetime are consistent with regulatory criteria identified in SRP Section 4.2.

The staff conducted two separate audits at the GEH offices in Wilmington, NC. Pacific Northwest National Laboratory staff assisted the agency staff in the review of the ESBWR

Marathon CRB structural evaluations and participated in the audits. A summary of the audits is given in the NRC's "Audit Report and Summary (2007 & 2008) for Global Nuclear Fuels Control Blade and Fuel Assembly Design," dated December 22, 2008 (Reference 10).

3.1 <u>ESBWR Marathon Control Rod Mechanical Design Evaluation</u>

This section evaluates the ESBWR Marathon CRB mechanical analysis results contained in Reference 1. GEH currently manufactures the long-life Marathon CRB for BWR/2 through BWR/6. The NRC acceptance of the Marathon CRB is documented in a letter dated July 1, 1991 (Reference 11). The design presented in Reference 1 is a Marathon CRB adapted for use in the ESBWR.

3.1.1 Design Change Description

The GEH ESBWR Marathon CRB is a derivative of the boiling-water reactor (BWR)/2–6 Marathon design approved in NEDE-31758P-A, "GE [H] Marathon Control Rod Assembly," issued October 1991 (Reference 8). The primary difference between the ESBWR Marathon and the BWR/2–6 Marathon designs is a shorter absorber section appropriate for ESBWR application. The ESBWR Marathon design uses the same square absorber tube design as the BWR/2–6 Marathon design. Compared to the original BWR/2–6 design, the ESBWR capsule of boron carbide (B_4C) has [[

]]

The basic design of the ESBWR Marathon CRB is the same as that of the BWR/2–6 Marathon CRB approved in the July 1, 1991 US NRC letter (Reference 11). The ESBWR Marathon CRB is an all B_4C design, with all tubes filled with either B_4C capsules or empty capsule plenums. The six design changes to the BWR/2–6 Marathon are listed below:

(1) <u>Absorber Section Length</u>: Absorber zone length for the ESBWR Marathon CRB assembly is shorter than in the BWR/2–6 since the active fuel height of the ESBWR design is shorter than the BWR/2–6 active fuel height. The nominal length of the BWR/2-6 Marathon CRB is [[ ]]; for the ESBWR Marathon CRB, this is reduced to [[ ]] (Table 2-1 of Reference 1).

(2) <u>Capsule Geometry</u>: The ESBWR Marathon CRB uses a B_4C capsule with the same cross-sectional dimensions as the Marathon-5S CRB capsule (NEDE-33284P, Revision 1, "GEH Proprietary Information Licensing Topical Report 'Marathon-5S Control Rod Assembly,'" issued November 2007 (Reference 12)). However, the ESBWR CRB design uses [[ ]].

The boron neutron absorption reaction releases helium gas, some of which is retained within the boron carbide powder matrix, causing the powder column to swell. [[

]]

[[

]]

(3) Capsule Length: The ESBWR Marathon CRB absorber section is shorter than that of the new BWR/6 Marathon control rod assembly. The nominal capsule length of the original Marathon CRB for BWR/2–6 was 11.4 inches. The absorber section design nominal lengths for the ESBWR Marathon CRB are [[ ]] and [[ ]]. These lengths are shorter than the nominal lengths of the new BWR/2–6 Marathon-5S CRB [[ ]].

Per NEDE-33244P, Rev. 1 (Reference 1), [[

]]

(4) Connector: The ESBWR control rod drive system uses an electrohydraulic fine motion control rod drive mechanism that provides electric-motor-driven positioning for normal insertion, and a hydraulic control unit provides rapid insertion (scram) during abnormal operating conditions. To be compatible with the fine motion control rod drive mechanism, the ESBWR Marathon CRB uses a connector as shown in Figures 2-3 and 2-4 of Reference 1, rather than the velocity limiter used in Marathon CRBs for BWR/2–6.

(5) Handle with Spacer Pads: The ESBWR Marathon CRB uses a raised spacer pad similar to that currently used for the D-lattice (BWR/2–4) Marathon CRB applications. The raised spacer pad is expected to eliminate the possibility of stress-corrosion cracking (SCC). The original Marathon CRB uses a traditional handle with rollers with wear pads that is susceptible to SCC within its handle pin-hole.

(6) Full-Length Tie Rod: The Marathon CRB uses multiple tie rod segments along the center of the cruciform shape. The Marathon-5S CRB uses a single tie rod segment along the center of the cruciform shape that runs the entire length of the assembly.

The ESBWR Marathon CRB employs a single tie rod that runs through the entire length of the assembly, similar to that in the Marathon-5S CRB.

3.1.2 Mechanical Design Evaluation

The review methodologies used in the evaluation of acceptability of the ESBWR Marathon CRB design are the following (References 1, 8, and 9):

1. Control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection.

2. The control rod design shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.

3. Control rod materials shall be shown to be compatible with the reactor environment.

4. Control rod reactivity worth shall be included in the plant core analyses.

5. The ESBWR Marathon CRB surveillance program shall be shown to provide increased assurance of the CRB's ability to perform their intended functions.

This section will discuss the first three licensing criteria. Section 3.2 will address Criterion 4, and Section 3.3 will address the proposed surveillance program.

3.1.2.1 <u>Stress, Strain, and Fatigue</u>

GEH's proposed licensing criterion is that control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection due to normal, abnormal, and faulted loads. The integrity of the welds under these loading conditions is also part of this criterion. This criterion is consistent with the SRP Section 4.2 (Reference 7) acceptance criteria for complying with the agency's regulations.

Section 3 of Reference 5 describes the method of analysis for the structural evaluation of the ESBWR Marathon CRB under various loading conditions including the worst-case or bounding loads and limiting material properties.

Effective stresses and strains were determined using the Von Mises distortion energy theory and compared to allowable limits. Both the Von Mises and Tresca stress criteria are used to predict the conditions for yielding under both uniaxial and multiaxial stress states. The Tresca criterion measures the maximum shear stress present, and the Von Mises criterion accounts for all principal stresses in the calculation of the conditions where yielding occurs.

Each structural analysis is first evaluated to determine whether unirradiated or irradiated material properties are appropriate. The analyses are divided into two categories: (1) analysis with an applied load (i.e., scram) for which a maximum stress is calculated and compared to the limiting unirradiated stress limit, and (2) analysis with an applied displacement (i.e., seismic bending) for which a maximum strain is calculated and compared to the limiting irradiated strain.

The evaluation used the licensing criteria of Section 4 of Reference 1, which states that control rod stresses and strains and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material, structure, or welded connection. The figure of merit employed by GEH for the stress-strain limit is the design ratio, which is effective stress divided by stress limit or effective strain divided by strain limit. The design ratio must be less than or equal to 1.0. The evaluation adds conservatism by limiting stresses to one-half of the ultimate tensile value (Table 3-2 of Reference 1).

The irradiation-resistant special melt austenitic stainless steel type 304 (304S) used for CRB absorber tubes is manufactured using standard industrial processes and solution annealing. Results from GEH's irradiated tensile tests demonstrated the strength and ductility of the material in the radial direction. As part of an audit (Reference 10), the staff investigated the ductility limit of 304S at end-of-life (EOL) properties and the accuracy of the stress-strain curves used to model the post yield behavior at beginning-of-life (BOL) and EOL. In Request for

]]

Additional Information (RAI) 4.2-32, the staff requested experimental data to validate GEH finite-element analysis (FEA) modeling assumptions. Comparison of the as-modeled stress-strain curves to actual test data showed different behavior past the material yield point. This was not critical for irradiated cases because none of the irradiated analyses had loads beyond the yield point; however, the unirradiated cases were of concern. To address these concerns, GEH demonstrated that the as-modeled stress-strain curves were greatly conservative compared to actual 304S test data as documented in the response to RAI 4.2-32. Based on the applicant's response, RAI 4.2-32 was resolved.

During the review of the FEA modeling approach, the staff also noted that the worst-case geometry was not being modeled.

In RAI 4.2-31 (Reference 2), the staff asked GEH to demonstrate that adequate design margin still existed by using the as-modeled geometry to perform a worst-case geometry check. The worst-case geometry evaluation showed that substantial design margin still remained. Based on the applicant's response, RAI 4.2-31 was resolved.

Since the ESBWR Marathon CRB uses the same square absorber tube as the BWR/2–6 Marathon CRB, welding processes are the same. Welding processes for CRBs are developed and qualified against a set of acceptance standards, which includes meeting minimum penetration standards, smooth blends between blended numbers, and no cracks, holes, or porosity.

In NEDE-33244 (Reference 1), GEH commits to performing weekly metallographic evaluations on sample laser welds to confirm that the results of the welding process remain acceptable. Figure 3-16 of Reference 1 shows photomicrographs of a typical laser weld. Comparison of the grain structure at the edge of the weld to an area away from the weld shows that there is no effective heat-affected zone (HAZ) for a laser weld. This combined lack of HAZ, Ta stabilization, and low carbon chemistry accounts for the good carbide test results.

The combination of low heat input welding, Ta stabilization, and restrictive carbon limits provides an effective barrier to intergranular SCC.

The CRB absorber tubes and capsules must provide positioning and containment of the neutron absorber material throughout its nuclear and mechanical life and prevent migration of the absorber material out of the capsules during normal, abnormal, emergency, and faulted conditions. The ESBWR Marathon CRB contains capsules with B_4C powder contained within absorber tubes. Helium gas produced from the boron-10 (B-10) (n, α) reaction causes the B_4C powder to swell, which causes the capsule to expand. A fraction of the helium escapes from the capsule and fills the absorber tube gap. In the BWR/2–6 Marathon capsule design, the expansion of the capsule causes a strain in the absorber tube near the end of nuclear life. Per NEDE-33244 (Reference 1) and RAI 4.2-27 response (Reference 4), the ESBWR capsule is designed such that its expansion allows for clearance between the capsule and the absorber tube, even at 100-percent local B_4C depletion at all locations along the length of the tube.

Test capsules of B_4C were irradiated in a reactor for a period of 10 years, and their dimensions were measured before and after irradiation in a hot cell. Diametral swelling results based on a conservative [[

]]

A thermal FEA of the absorber tube was performed to determine the peak B_4C temperature using worst-case dimensions, maximum EOL crud buildup, combined with maximum BOL heat generation. The pressure in the absorber tube results from the helium release that increases with temperature. For the thermal FEA analysis, corrosion was modeled as the buildup of an insulating layer of crud, the thickness of which was twice that assumed for the BWR/2–6 Marathon design. Data from the analysis (Table 3-10 of Reference 1) shows significant dependence of helium release fraction on the irradiation temperature.

To assess the limiting mechanical lifetime for the ESBWR Marathon CRB, the staff performed confirmatory FEAs to determine the pressurization capability of the absorber tube due to helium gas production.

The burst pressure is defined as the internal pressure at which any point in the tube reaches stress intensity equal to the true ultimate strength of the material. The burst pressure capability of the tube is calculated using square absorber tube nominal dimensions. The applicant reduced the burst pressure by half to add a safety factor of 2 to the peak operating pressure. The resulting peak operating pressure is further reduced (scaled down) by [[ ]], based on experimental observations compared against FEA results of a previous absorber tube design. Table 3-20 of NEDE-33244 (Reference 1) indicates that although the burst pressure is less than the nominal single tube value, it is bounded by the scaled burst pressure. Therefore, the staff concludes that the design-basis absorber tube allowable pressure is conservative.

Analysis of stress components performed by GEH at the point of maximum stress intensity for the absorber tube, with maximum allowable internal pressure, found the point of maximum stress intensity on the inside of the absorber tube. The principal stress components as listed in Table 3-18 of NEDE-33244 (Reference 1) are within the allowable stress value for 304S tubing.

Welding of the absorber section of the CRB generates two potential issues: sensitization and residual stress. The low-heat-input laser welding process does not result in sensitized material. The integrity of the absorber tube welds is confirmed by the weekly metallographic evaluations performed by GEH as specified in NEDE-33244 (Reference 1).

One major effect of the welding process is that it will introduce tensile residual stress in the narrow weld/HAZ region. Residual stresses are not a significant concern for two reasons: (1) the field cracking has not been associated with the weld HAZ and (2) the irradiation experienced by the CRB over the initial time of operation can significantly reduce these stresses by 60 percent or more through the radiation creep process. At the level of reduced stress, there is little concern for any effect on SCC initiation.

Pressurization of the absorber tube causes axial expansion of the tubes as the internal pressure pushes against the end plugs that seal the ends of the absorber tubes. Table 3-19 of Reference 1 lists the maximum elongation and other parameters used in the calculation. The maximum elongation is found to be relatively small and will not affect the fit, form, or function of the CRB.

The finite-element model for pressurization calculations uses unirradiated material properties. To confirm that usage of unirradiated material properties is conservative, GEH performed two test cases using irradiated material properties and calculated peak stress and strain intensities and compared them to material allowable ultimate stress and strain in Table 3-21 of Reference 1. This comparison confirmed that the unirradiated material analysis for absorber tube pressurization is generally conservative. The staff reviewed the material property comparisons, and agrees that the use of unirradiated material properties is more conservative for use as input to this finite-element model.

The original Marathon CRB (described in Reference 8) possessed a tendency for irradiation-assisted stress-corrosion cracking (IASCC) in the handle near the roller pin. To eliminate the possibility of IASCC initiating within the handle pin-hole, the ESBWR Marathon CRB employs a raised spacer pad, shown in Figure 2-4 of Reference 1. Also, the Marathon absorber tube is manufactured from GEH proprietary stainless steel, Rad Resist 304S, which is optimized to be resistant to IASCC.

The ESBWR Marathon CRB, in addition to using IASCC-resistant 304S, is designed such that the swelling of the B_4C capsule does not exert a mechanically imposed stress or strain on the absorber tube, thereby reducing the likelihood of SCC.

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The applicant performed an absorber tube mechanical limit calculation as a function of average percent depletion of B-10 based on peak heat generation, temperatures, and helium release fractions. Table 3-23 of NEDE-33244 (Reference 1) lists the CRB mechanical lifetime, in terms of four-segment average B-10 depletion. The four-segment mechanical lifetime limit is [[ ]]. The staff concurs with this mechanical lifetime limit based on the use of approved methods and conservative assumptions. The nuclear depletion limit is calculated to be [[ ]] in NEDE-33243 (Reference 6). Therefore, the nuclear lifetime of the ESBWR Marathon CRB is limiting, in the sense that the mechanical lifetime exceeds the nuclear lifetime.

Section 3.8 of Reference 1 details the load combinations, including AOOs and fatigue loads associated with those combinations. Loads experienced by the CRB during scram and safe-shutdown earthquake (SSE) were evaluated. The combined loads were evaluated for the cumulative effect of maximum cyclic loadings, based on reactor cycles. The fatigue usage was evaluated against a limit of 1.0. Tables 3-12 through 3-14 of Reference 1 show that the low number of cycles represents only a small amount of cumulative damage, well below the design limit. This conclusion is based on the ASME Code Section III fatigue design for unirradiated austenitic material.

Fatigue evaluation for any flow-induced vibration for the CRB was performed in terms of loads induced by transverse loading. It was determined that the load considered was so small that it would not lead to any risk of fatigue.

Based on the use of conservative material properties, conservative design approach, review of GEH calculations, and staff confirmatory analyses, the staff finds the ESBWR Marathon CRB mechanical design analyses acceptable.

3.1.2.2 <u>Control Rod Insertion</u>

SRP Section 4.2 acceptance criteria stipulate that the control rod reactivity and insertability must be maintained. The GEH proposed licensing criterion stating that the CRB design shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses is consistent with SRP Section 4.2 and therefore acceptable.

Conservative structural analyses of the CRB were performed assuming 100-percent failed control rod drive buffer. Structural stresses were determined from scram loads, using weld quality factors and worst-case geometry (Figure 3-1 and Table 3-4 of Reference 1). Table 3-5 of Reference 1 indicates that sufficient margin exists against failure for all cross-sections and welds.

Section 3.4 of Reference 1 addresses CRB insertion with respect to seismic conditions and fuel channel bow-induced bending. The SSE analysis for the ESBWR Marathon CRB was performed by evaluating the strain in the CRB absorber section with maximum SSE deflection. Table 3-6 of Reference 1 shows that under combined worst-case conditions, the maximum strain is well below the limiting maximum allowable strain at irradiated conditions.

FEA evaluating the combined effect of CRB bending due to SSE and channel bulge and bow deflection at the full-length tie rod to absorber tube weld showed that the resulting stresses are acceptable against design criteria (Figure 3-2 and Table 3-7 of Reference 1).

FEA to determine the effect of lateral load imposed on CRB absorber due to an excessively bowed channel indicated that the maximum stress intensity is less than the absorber tube allowable load (Table 3-12 of Reference 1). This lateral load model was also evaluated using EOL irradiated material properties; the evaluation determined that both maximum stress intensity and maximum strain intensity were less than corresponding ultimate stress and strain, respectively.

Based on the staff's review of the underlying engineering calculations and results presented in Reference 1, the staff finds that the ESBWR Marathon CRB design satisfies the control rod insertion licensing criterion.

3.1.2.3 Control Rod Material

GEH's proposed licensing criterion is that the material of the CRB shall be shown to be compatible with the reactor environment for the life of the CRB. The CRB design shall consider the effects of crudding, crevices, stress corrosion, and irradiation on the material. This criterion is consistent with SRP Section 4.2 and therefore acceptable.

The ESBWR Marathon CRB uses the same materials as the current Marathon design. No new material has been introduced. The square absorber tubes are made of the same high-purity stabilized type 304S. Table 1 of Reference 1 lists limiting unirradiated material strengths of the CRB components. GEH requires that the mechanical properties of all materials used in the fabrication of CRBs meet material specification limits.

Based on the information submitted, the staff finds that the ESBWR Marathon CRB design has satisfied the licensing criterion.

3.2 ESBWR Marathon Control Rod Nuclear Design Evaluation

3.2.1 Design Specifications

Section 1 of GEH letter MFN 08-562, dated July 3, 2008 (Reference 6), describes the ESBWR Marathon CRB nuclear design specifications. The ESBWR Marathon CRB is designed such that its initial cold worth matches the Duralife CRB initial cold worth. The CRB EOL is specified as the percent B-10 depletion resulting in a 10-percent cold worth reduction in any quarter segment of the blade. This EOL specification is consistent with previous CRB designs. GEH Letter MFN 09-343, dated June 11, 2009 (Reference 13) gives the historical basis for the nuclear lifetime criterion.

3.2.2 Nuclear Design Evaluation

3.2.2.1 Depletion Methodology

The nuclear lifetime for a particular CRB is calculated by the use of a two-dimensional Monte Carlo analysis applied in a stepwise fashion to account for B-10 depletion over time. For each time step, the poison reaction rates are assumed to be constant, and the poison inventories are calculated in each discrete area of the blade. The poison number densities are then updated by averaging on a cell-by-cell basis, and the process is repeated until the reduction in cold worth reaches the end-of-nuclear-life criterion.

This process was used and approved previously for both the Marathon and Marathon-5S CRB designs (References 8 and 12, respectively). For the ESBWR nuclear design evaluations, GEH has replaced the Monte Carlo code used in Reference 8 (MERIT) with a GEH-controlled version of MCNP4A. MCNP4A is an industry standard code developed by Los Alamos National Laboratory (LANL) for Monte Carlo analysis of neutron transport. It has been approved by the staff for use for many different uses for many years. The staff therefore approves this change in methodology.

As previously noted in Reference 4, the NRC staff is concerned about system geometry definitions for the poison region when a MCNP-based approach is used to calculate B-10 depletion. To preclude the appearance of B-10 drifting during the averaging of number densities in each region for each time step, the region thicknesses should be on the order of one neutron mean free path. Reference 6 accounts for B-10 drift by dividing the B_4C column into four rings of equal area, resulting in ring thicknesses that are approximately one neutron mean free path or less. The staff finds that by dividing the B_4C column in this fashion, the averaging of the poison number densities over the cell at each time step is a reasonable approximation of how the B-10 poison would be burned out over time and is therefore acceptable.

3.2.2.2 Nuclear Lifetime and Initial Control Rod Worth

The NRC staff reviewed the ESBWR CRB nuclear lifetime results for the lattice provided in the MCNP4A input deck (GEH Letter MFN 08-464, dated May 27, 2008 (Reference 14)), as calculated by the methodology described in Section 3.2.2.1 of this safety evaluation. Based on the findings in Section 3 of Reference 6, the EOL fluence is [[ ]] (corresponding to [[ ]] maximum quarter segment B-10 depletion). Using Reference 14, the NRC staff was able to verify that this EOL fluence is appropriate through confirmatory analyses involving conservative assumptions. Table 2.1 of Reference 6 provides the nominal and limiting axial burnup profiles used to determine the blade EOL. Furthermore, Section 5 of Reference 6 describes how the axial depletion profile is used in conjunction with the radial depletion profile to determine the average depletion fraction in the limiting rod. This is important for determining if the depletion is within the limit of the mechanical design depletion limit. Since results show that the depletion in the limiting tube is greater than the mechanical limit, a plenum region at the bottom is introduced at the bottom of the tube for pressure relief from the helium release.

During audits conducted at GEH's Wilmington, NC, site, the NRC staff reviewed sample MCNP decks used in calculating the CRB reactivity worth (Reference 10). The staff determined that the methodology described in Section 3.2.2.1 of this safety evaluation was correctly implemented. Specifically, the staff inspected the MCNP input decks to ensure that (1) the B_4C regions were modeled to handle "B-10 drift", (2) material densities were correctly calculated, (3) the overall geometry matched that presented in Reference 6, and (4) temperature-dependent neutron cross-section libraries were used appropriately and concurrently with material temperature specification; this includes $S(\alpha,\beta)$ cross-section libraries.

Per NEDO-33244 P-A, "Licensing Topical Report, ESBWR Marathon Control Rod Mechanical Design Report," GEH has committed to bounding the effects of the reduction in control blade worth by imposing a 1% shut down margin design criteria which is nearly 3 times greater than the tech spec limit in the analyses. In addition, the startup physics testing confirms the blade worths. Based on the additional margin used in the analyses and the physical testing performed on a cycle by cycle basis, the staff concludes that there is reasonable assurance that all safety limits related to the nuclear lifetime and shut down margin will be met.

3.2.2.3 Heat Generation Rates

One major concern in calculating the CRB mechanical lifetime is the calculation of the internal pressure resulting from the release of helium generated after a B-10 atom captures a neutron. Helium release is directly affected by temperature, and therefore the heating rate is critical when calculating mechanical lifetime.

GEH calculates the heat generated by the neutron-CRB interaction as solely an (n,α) interaction resulting in 2.79 megaelectronvolts (MeV) of energy deposition for each interaction between a neutron and a B-10 atom. The NRC staff was concerned about a possible undercounting of the energy deposition because the carbon-neutron scattering and gamma contributions were not being considered. In RAI 4.2-30 (Reference 15), the staff requested additional information regarding GEH's handling of energy deposition within the MCNP models.

GEH explained (Reference 15) that by calculating the μ value (ratio of average absorptions in the control poison to the total fissions in the adjacent bundles) and multiplying by the 2.79-MeV energy deposition, the calculational method used in Reference 6 is bounding. On average, 2.79 MeV is deposited locally in the B_4C absorber region only 6 percent of the time, while 94 percent of the time, interactions between a neutron and B-10 produce lithium-7 (Li-7) in an excited state, which then produces a 0.48-MeV (17 percent of the total 2.79 MeV) gamma ray through decay. The NRC staff performed a confirmatory calculation to determine the mean free path of a 0.48-MeV photon traveling through compacted B_4C powder (70 percent theoretical density). Using a mass attenuation coefficient of 9.55×10^{-2} square centimeters per gram (cm^2/g) (based on carbon, which is more limiting than boron) and a density of 1.76 grams per cubic centimeter (g/cm^3), the linear attenuation coefficient is 0.168 cm^{-1}, resulting in a mean free path of approximately 6 cm. This shows that the probability of interaction of the decay gamma in the B_4C absorber region is negligible, which implies that on average, approximately 2.34 MeV is deposited in the B_4C region per B-10 neutron absorption. By assuming that the entire 2.79 MeV remains in the absorber region, the GEH calculation is considered bounding. GEH calculations show this in a confirmatory analysis (GEH Letter MFN 08-840, dated October 31, 2008 (Reference 15)), which takes a more realistic approach by including all reaction types in the B_4C absorber region (including elastic scattering with carbon atoms and all photon interactions). The assumption in Reference 6 of 2.79 MeV per absorption results in a heating rate that is approximately [[ ]] higher than GEH's realistic confirmatory case. The staff agrees that

GEH's assumptions in its simplified heating rate calculations, as described in Reference 6, are conservative and appropriate.

3.3 ESBWR Marathon Surveillance Program

During staff audits at GEH-Wilmington (described in References 10 and 16), the NRC staff informed GEH that a rigorous surveillance program should be described for ESBWR Marathon CRBs consistent with the approved surveillance program for Marathon-5S. The staff discussed this concern in RAI 4.2-26 S01. In response, GEH proposed a revision to section 5.0 of LTR NEDE-33244P to specify surveillance requirements for ESBWR Marathon CRBs. The staff determined that the surveillance requirements, as followed, would detect any unusual degradation before a safety significant issue could develop and therefore finds the proposed requirements to be acceptable for inclusion in the approved version of the LTR. Based on the applicant's response, RAI 4.2-26 was resolved.

4.0 CONCLUSION

Based on its review of NEDE-33243P, Revision 2, NEDE-33244P, Revision 1, and the conclusion that the CRB design meets the criteria described in Section 2 of this report, the staff finds the ESBWR Marathon CRB design acceptable for licensing applications in ESBWR power plants. Licensees referencing this LTR will need to comply with the conditions listed in Section 5.0.

5.0 CONDITIONS AND LIMITATIONS

Applicants referencing NEDE-33243P, Revision 2, and NEDE-33244P, Revision 1, must ensure compliance with the following conditions and limitations:

(1) The ESBWR Marathon CRB design is restricted to the design specifications provided in Sections 2 and 3 of References 1 and 6. Changes in component design, materials, or processing specifications may alter the in-reactor behavior of this design and the basis of the staff's approval.

(2) The ESBWR Marathon CRB design is restricted to the use of B_4C absorber material. The introduction of alternative absorber materials (e.g., hafnium) requires NRC review and approval. Further, the staff's review did not consider enriched B_4C powder (i.e., powder with an artificially increased B-10 isotopic concentration); therefore, this safety evaluation does not approve the use of enriched powder.

(3) The inspection and reporting requirements in the ESBWR Marathon Surveillance Program, detailed in the response to RAI 4.2-26 S01 (Reference 2), must be fulfilled.

6.0 REFERENCES

1. GE Hitachi Nuclear Energy, Letter MFN 07-612, "Submittal of Licensing Topical Report NEDE-33244P, Revision 1, 'ESBWR Marathon Control Rod Mechanical Design Report,'" November 15, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML073240196, ML073240227, ML073240220); Revision 2 Mark-ups, January 28, 2010 (ADAMS Accession No. ML100320352).

2. GE Hitachi Nuclear Energy, Letter MFN 08-757, "Response to Portion of NRC Request for Additional Information Letter No. 243—Related to ESBWR Design Certification Application—RAI Numbers 4.2-24 Supplement 1, 4.2-26 Supplement 1, 4.2-31 and 4.2-32," October 8, 2008 (ADAMS Accession Nos. ML082880089, ML082880090).

3. GE Hitachi Nuclear Energy, Letter MFN 08-527, "Response to Portion of NRC Request for Additional Information Letter No. 204—Related to NEDE-33244P, Revision 1, 'ESBWR Control Rod Mechanical Design Report, RAI Number 4.2-20,'" June 27, 2008 (ADAMS Accession Nos. ML081830148, ML081830149).

4. GE Hitachi Nuclear Energy, Letter MFN 08-474, "Response to Portion of NRC Request for Additional Information Letter No. 167—Related to NEDE-33244P, Revision 1 'ESBWR Control Rod Mechanical Design Report, RAI Number 4.2-21 through 4.2-27,'" June 13, 2008 (ADAMS Accession Nos. ML081690506, ML081690507).

5. GE Hitachi Nuclear Energy, NEDE-33244P, Revision 1, "Licensing Topical Report, ESBWR Marathon Control Rod Mechanical Design Report" November 2007 (ADAMS Accession Nos. ML073240220, ML073240196, ML073240227).

6. GE Hitachi Nuclear Energy, "Licensing Topical Report ESBWR Control Rod Nuclear Design," NEDE-33243P, Revision 2, July 2008 (ADAMS Accession Nos. ML081980190, ML081980189, ML081980192).

7. U.S. Nuclear Regulatory Commission, NUREG-0800, "U.S. NRC Standard Review Plan," Section 4.2, "Fuel System Design," Revision 3, March 2007 (ADAMS Accession Nos. ML070740002).

8. GE Hitachi Nuclear Energy, "GE[H] Marathon Control Rod Assembly," NEDE-31758P-A, GEH Proprietary, October 1991 (ADAMS Accession No. 9106280113).

9. GE Hitachi Nuclear Energy, 26A6642A, Revision 5, "ESBWR Design Control Document," Tier 2, Chapter 4, "Reactor," May 2008 (ADAMS Accession No. ML081820527).

10. U.S. Nuclear Regulatory Commission, Memorandum, "Audit Report and Summary (2007 & 2008) for Global Nuclear Fuels Control Blade and Fuel Assembly Design," December 22, 2008 (ADAMS Accession No. ML083230072).

11. U.S. Nuclear Regulatory Commission, Letter, "Acceptance for Referencing of Topical Report NEDE-31758P, 'GE[H] Marathon Control Rod Assembly,'" July 1, 1991 (ADAMS Accession No. 9107090009).

12. GE Hitachi Nuclear Energy, NEDE-33284P, Revision 1, Class III DRF-0000-0036-8667, "GEH Proprietary Information Licensing Topical Report 'Marathon-5S Control Rod Assembly,'" November 2007 (ADAMS Accession Nos. ML073320285, ML073320294, ML073320297).

13. GE Hitachi Nuclear Energy, Letter to NRC, MFN 09-343, "Draft Safety Evaluation by the Office of Nuclear Reactor Regulation Licensing Topical Report (LTR) NEDE-33284,

Revision 1, 'Marathon-5S Control Rod Assembly,'" June 11, 2009 (ADAMS Accession No. ML091630582).

14. GE Hitachi Nuclear Energy, Letter MFN 08-464, "Response to Portion of NRC Request for Additional Information Letter No. 162 Related to NEDE-33243P, Revision 1 'ESBWR Control Rod Nuclear Design' RAI Numbers 4.2-16, 4.2-17, 4.2-18, 4.2-19. ESBWR Lattice 1803 M5 Blade Calculation," May 27, 2008 (ADAMS Accession Nos. ML0817101628, ML081710127, ML081710128, ML081930949).

15. GE Hitachi Nuclear Energy, Letter MFN 08-840, "Response to Portion of NRC Request for Additional Information Letter No. 243—Related to Design Control Document Revision 5—RAI Number 4.2-30," October 31, 2008 (ADAMS Accession Nos. ML083090098, ML083090099).

16. U.S. Nuclear Regulatory Commission, Memorandum, "Audit Report for ESBWR GE14E Fuel Assembly Design, ESBWR Marathon Control Blade Design, and Marathon-5S Control Blade Design," September 11, 2007 (ADAMS Accession No. ML072550006).

17. *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria."

18. *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities."

Enclosure 3

MFN 10-289

Affidavit

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Mark J. Colby, state as follows:

(1) I am the Manager, New Plants Engineering of GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of GEH's letter, MFN 10-289, Mr. Richard E. Kingston to U.S. Nuclear Regulatory Commission, entitled *Submittal of Accepted Versions of NEDE-33244P, "ESBWR Marathon Control Rod Mechanical Design Report,"* dated September 29, 2010. GEH text proprietary information in Enclosure 1, which is entitled *NEDE-33244P-A, Revision 2, "ESBWR Marathon Control Rod Mechanical Design Report"* is identified by a dark red dotted underline inside double square brackets. [[This sentence is an example.{3}]] Figures and large equation objects containing GEH proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination. Note that the GEH proprietary information in the NRC's Final Safety Evaluation, which is enclosed in NEDE-33244P-A, Revision 2, is identified with red text inside double square brackets. [[This sentence is an example.]]

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act (FOIA), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH and/or other companies.

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains detailed design, methodology, and dimensional information regarding the ESBWR Marathon Control Rod developed by GEH over a period of several years at a substantial cost.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply

the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 29th day of September 2010.

Mark J. Colby

Mark J. Colby
GE-Hitachi Nuclear Energy Americas LLC