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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 67 Related to ESBWR Design Certification Application – DCD Section 3.9 – RAI Numbers 3.9-4 through 3.9-11, 3.9-17, 3.9-18, 3.9-23, 3.9-26, 3.9-27, 3.9-29, 3.9-32, 3.9-34 through 3.9-36, 3.9-38 through 3.9-40, 3.9-44, 3.9-46 through 3.9-55, 3.9-57, 3.9-59, 3.9-60, 3.9-67, 3.9-72 through 3.9-76, 3.9-79, 3.9-80, 3.9-91 through 3.9-94, 3.9-96 through 3.9-99, 3.9-101, 3.9-102, 3.9-104, 3.9-105, 3.9-108, 3.9-110, 3.9-132, 3.9-140, 3.9-142, 3.9-147, 3.9-150, 3.9-151, and 3.9-153**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

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

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

D068

Reference:

1. MFN 06-378, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 67 Related to ESBWR Design Certification Application,* October 10, 2006

Enclosure:

1. MFN 06-464 – Response to Portion of NRC Request for Additional Information Letter No. 67 Related to ESBWR Design Certification Application – DCD Section 3.9 – RAI Numbers 3.9-4 through 3.9-11, 3.9-17, 3.9-18, 3.9-23, 3.9-26, 3.9-27, 3.9-29, 3.9-32, 3.9-34 through 3.9-36, 3.9-38 through 3.9-40, 3.9-44, 3.9-46 through 3.9-55, 3.9-57, 3.9-59, 3.9-60, 3.9-67, 3.9-72 through 3.9-76, 3.9-79, 3.9-80, 3.9-91 through 3.9-94, 3.9-96 through 3.9-99, 3.9-101, 3.9-102, 3.9-104, 3.9-105, 3.9-108, 3.9-110, 3.9-132, 3.9-140, 3.9-142, 3.9-147, 3.9-150, 3.9-151, and 3.9-153

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRFs	0061-0794, 0061-5824, 0060-8483, 0061-6792, 0061-5812

Enclosure 1

MFN 06-464

Response to Portion of NRC Request for

Additional Information Letter No. 67

Related to ESBWR Design Certification Application

DCD Section 3.9

RAI Numbers 3.9-4 through 3.9-11, 3.9-17, 3.9-18, 3.9-23, 3.9-26, 3.9-27, 3.9-29, 3.9-32, 3.9-34 through 3.9-36, 3.9-38 through 3.9-40, 3.9-44, 3.9-46 through 3.9-55, 3.9-57, 3.9-59, 3.9-60, 3.9-67, 3.9-72 through 3.9-76, 3.9-79, 3.9-80, 3.9-91 through 3.9-94, 3.9-96 through 3.9-99, 3.9-101, 3.9-102, 3.9-104, 3.9-105, 3.9-108, 3.9-110, 3.9-132, 3.9-140, 3.9-142, 3.9-147, 3.9-150, 3.9-151, and 3.9-153

NRC RAI 3.9-4

In DCD Tier 2, Table 3.9-1, provide clarification if the term "No. of Events" is synonymous with "cycles".

GE Response

The events 1 through 16 are listed in Table 3.9.1. The cycles are the number of anticipated operational events during the 60-year life of the reactor. The table column heading "No. of Events" will be changed to "No. of Cycles" and the dynamic loading events cycles will be revised as necessary.

DCD/LTR Impact

DCD Tier 2 Table 3.9-1 will be revised as noted in the attached markup.

NRC RAI 3.9-5

Discuss the basis for the Plant Operating Events and corresponding "number of events" listed in DCD Tier 2, Table 3.9-1.

GE Response

The plant operating events listed in Table 3.9-1 are anticipated events during the life of the reactor. The ESBWR events are basically the same as the events specified for all earlier BWRs including ABWR. The number of specified events over the 60-year reactor life is based on earlier BWR experience.

DCD/LTR Impact

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

NRC RAI 3.9-6

Discuss the difference between the Event 3, Startup, events (180) and Event 9, Shutdown, events (172) in DCD Table 3.9-1.

GE Response

In addition to the 172 shutdown events (event 9) specified, there are 8 Safety Relief Valve (SRV) or single Depressurization Valve (DPV) actuation events (event 15) specified after which the reactor is shut down. Hence, the number of 180 startup events.

DCD/LTR Impact

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

NRC RAI 3.9-7

Discuss the basis of the "Dynamic Loading Events" in DCD Tier 2, Table 3.9-1.

GE Response

The dynamic loading events apply the loads on reactor components calculated in the seismic and hydrodynamic analyses. These cyclic loads are included in the component fatigue analysis.

DCD/LTR Impact

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

NRC RAI 3.9-8

Discuss the basis of "2 events/10 cycles per event" for Event 13 in DCD Tier 2, Table 3.9-1.

GE Response

Number of earthquake cycles as defined in Table 3.9-1 is applicable to plants without an explicit OBE design consideration. This is modeled after the NRC-certified ABWR design. The same is adopted by a more recent AP1000 application. In Section 3.12.5.14 of the NRC FSER for AP1000, the following is stated:

"An acceptable cyclic load basis for fatigue evaluation is two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). Alternately, a number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D to IEEE Std 344-1987.".

DCD/LTR Impact

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

NRC RAI 3.9-9

Discuss the basis for selecting 1 cycle for Event 14 in DCD Tier 2, Table 3.9-1.

GE Response

Per subsection 3.9.3.1 the annual encounter probability of a single level D event is less than 10^{-4} or less than 10^{-2} over the 60 year plant life. Therefore, faulted event 14 of the DCD table is evaluated to happen 1 time.

DCD/LTR Impact

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

NRC RAI 3.9-10

Discuss the basis for selecting 1 event for Event 16 in DCD Tier 2, Table 3.9-1.

GE Response

Per subsection 3.9.3.1 the annual encounter probablility of a single level D event is less than 10^{-4} or less than 10^{-2} over the 60 year plant life. Therefore, faulted event 16 of the DCD table is evaluated to happen 1 time. .

DCD/LTR Impact

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

NRC RAI 3.9-11

Provide confirmation that the transients in DCD Tier 2, Table 3.9-1 are valid for 60-year operation.

GE Response

A statement will be added to DCD Tier 2 revision 3 that transients are for 60 years.

DCD/LTR Impact

DCD Tier 2 Table 3.9-1 will be revised as noted in the attached markup.

NRC RAI 3.9-17

Provide a listing of the high and moderate energy piping systems which are covered by the vibration and dynamic effects testing program described in the DCD Tier 2, Section 3.9.2.1.1 and those that are exempted from it. Also provide the bases for these exemptions.

GE Response

In accordance with RG 1.68 Appendix A the following systems or portion of system are covered by the vibration and dynamic effects testing program:

- ·ASME Code Class 1, 2, and 3 Systems,
- ·Other High-Energy piping inside Seismic Category I structures
- ·High-Energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and
- ·Seismic Category I portions of moderate-energy piping systems located outside Containment.

The systems to be considered are the following:

- Nuclear Boiler System (B21)
- Isolation Condenser System (B32)
- Control Rod Drive System (C12)
- Standby Liquid Control System (C41)
- Gravity Driven Cooling System (E50)
- Fuel and Auxiliary Pools Cooling System (G21)
- Reactor Water Cleanup/ Shutdown Cooling System (G31)
- Fire Protection System (U43)
- Equipment and Floor Drain System (U50)
- Chilled Water System (P25)

DCD Impact

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

NRC RAI 3.9-18

In DCD Tier 2 Section 3.9.2.1.1, it is stated that there are essentially three methods available for determining the acceptability of steady state and transient vibration for the affected systems. These are visual observation, local measurements, and remotely monitored/recorded measurements. The technique used depends on a number of factors stated in the application. The staff finds this information inadequate to determine which specific measurement technique will be used for a particular system. Provide a listing of the systems to identify which measurement technique (visual observation, local measurements, or remotely monitored/recorded measurements) will be used on each of the piping system covered by the vibration and dynamic effects testing program.

GE Response

Within each applicable vibration category (steady-state and transient) the piping will be classified into one of the three vibration monitoring groups according the criteria presented in paragraphs 3.1.1 and 3.1.2 of ASME OM S/G Part 3:

1) Vibration Monitoring Group 1 (VMG1) (Remote sophisticated monitoring devices and extensive data collection): Systems that exhibit a response not characterized by simple piping modes. The locations of the measurement points will be selected taking into account the maximum deformation in the modes of greatest mass participation. The following systems are VMG1.

   - Main steam piping and SRV discharging piping in the drywell

   - Feedwater piping inside the containment

2) Vibration Monitoring Group 2 (VMG2) (Local measurements): Systems that may exhibit significant vibration response based on past experience with similar systems or similar system operating conditions. As a general rule, vibration measurement points will be located at:

   - Pump intakes and discharges,

   - Devices that cause pressure drops, like flow restrictors, control valves, etc.

   - Quick-acting valves

   - Check valves

3) Vibration Monitoring Group 3 (VMG3) (Visual methods): Systems that are not expected to exhibit significant vibrational response based on past experience with similar systems or similar system operating conditions. The following measurement points fall within this group:

   - Drains and vents

   - Instrumentation pipings.- Pumps in parallel

   - Weld junctions

   - Sensitive equipment (valves, heat exchangers, pumps, etc)

In addition, systems that are inaccessible for visual observation or measurement by portable devices or as a result of adverse environmental effects during the conditions listed in the test specification shall be classified into either VMG1 or VMG2.

DCD Impact

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

NRC RAI 3.9-23

There is insufficient information in DCD Tier 2, Section 3.9.2.1.1, relative to the visual inspections and measurements. Therefore the applicant is requested to provide a list of selected locations in the piping system at which visual inspections and measurements will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak) or other appropriate criteria intended to be used to show that the stress and fatigue limits are within the design levels, should be provided.

GE Response

Visual inspections are performed on systems that are not expected to exhibit significant vibration response based on past experience with similar systems or similar system operating conditions.

All drain, vent systems, instrumentation piping 1" and under, are not expect to have significant vibration. These systems can use visual inspections.

Any system using visual inspection should have vibration so low, that it will meet the ASME OM-S/G-1990 standard, Part 3 Appendix D velocity screen criteria, 0.5 in/sec, when the system is monitored. In this case, it is not necessary to define the peak-to-peak displacement.

The above criteria and the other criteria specified in Subsection 3.9.2.1.1 are used to establish the visual inspection and measurement plan when the system piping details and analysis have been completed. See also response to RAI 3.9-18.

DCD Impact

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

NRC RAI 3.9-26

There is insufficient information in DCD Tier 2,Section 3.9.2.1.1 relative to the corrective restraints. In accordance with Standard Review Plan (SRP) 3.9.2,Draft Revision 3, April 1996, if vibration is noted beyond the acceptance levels, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If during the test, piping system restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should be performed to determine that the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations where large motion is predicted, a description should be provided to address the identified discrepancy. Provide detailed information relative to corrective restraints.

GE Response

Corrective restraints are added if the existing restraints are determined inadequate or damaged, depending on the vibration frequency range (low or high frequency).

Low frequency vibration can be adequately restrained through the addition of supports, preferably located near bends, heavy concentrated masses and piping discontinuities.

Vibration of vents, drains, bypass and instrument piping can be corrected by bracing the masses to the main pipe eliminating relative vibrations.

Spring Sway Struts can be used for controlling low frequency vibration problems. Straps with elastomeric elements and no gap can be also used for high frequency, located at points with high dynamic vibration susceptibility.

The snubber piston travel should not be affected by the vibration displacements. This is because the snubber travel is significantly larger than the vibration amplitude, which is normally less than 0.020 inches.

DCD Impact

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

NRC RAI 3.9-27

In DCD Tier 2, Section 3.9.2, sufficient information is not provided for the qualification of Seismic Category I cable tray and conduit supports. Section 3.10.3.2 provided only limited information for loadings that are used for their design and analysis. Provide a detailed discussion on the methods and criteria used for the design of the seismic Category I electrical raceway (cable trays, conduit and heating, ventilation and air conditioning (HVAC)) supports, including the applicable codes, standards, and specifications used for the design. Also explain how the design would conform to the requirements of SRP 3.7.3.

GE Response

Please refer to RAI 3.8-52 response submitted under MFN 06-298.

DCD Impact

DCD Tier 2 Subsections 3.8.4.1.6, 3.8.4.1.7, 3.9.2, 3.10.3.2, 9.4.1.3, 9.4.2.3, and 9.4.6.3 will be revised as noted in the attached markup.

NRC RAI 3.9-29

DCD Tier 2, Section 3.7.3.3.2, fourth bullet, states that locating a mass at a point where the maximum displacement is expected to occur would tend to lower the natural frequencies of the equipment. It states that it is conservative because the equipment frequencies are in the higher spectral range of the response spectra. While these statements may be generally true for a seismic excitation, which has a predominant frequency content around 2 to 10 Hz, it may not be true for the equipment response under hydrodynamic loads, which typically have much higher frequency content. Similar concerns apply also to the statements made for the case of live loads and variable support stiffness. Revise the above statements or provide justification for the statements made.

GE Response

The 4th bullet DCD Tier 2, Subsection 3.7.3.3.2 reads as follows:

- When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to lower the natural frequencies of the equipment because the equipment frequencies are in the higher spectral range of the response spectra. Similarly, in the case of live loads (mobile) and variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the lowest frequency content for the system. This ensures conservative dynamic loads, since the equipment frequencies are such that the floor spectra peak is in the lower frequency range. If not, the model is adjusted to give more conservative results.

The last sentence of the bullet is "If not, the model is adjusted to give more conservative results."

This sentence requires that, if the design engineer knows that the analysis model is not conservative due to lower equipment frequency, then proper improvement of the model is required.

DCD Impact

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

NRC RAI 3.9-32

DCD Tier 2, Section 3.7.3.8, states that to simulate the dynamic effects of the non-Category I systems attached to Seismic Category I systems, the attached non-Category I systems, up to the first anchor beyond the interface, are also designed in such a manner that during an earthquake of SSE intensity it does not cause a failure of the Seismic Category I system. Clarify that this designated first anchor is designed as a six-way restraint in the specific non-Category I system.

GE Response

To simulate the dynamic effects of the non-Category I systems attached to Seismic Category I systems, the attached non-Category I systems are analyzed to include dynamic effects, up to the first anchor beyond the interface. This anchor is either a six-way restraint anchor or for a distance such that there are at least two seismic restraints in each of the three orthogonal directions.

DCD Impact

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

NRC RAI 3.9-34

In DCD Tier 2, Section 3.7.3.12, for the effect of differential building movements, sufficient information is not provided for the analysis methodology of subsystems which are anchored and restrained to floors and walls of buildings that may experience relatively large differential displacements between separate buildings at a high seismic activity site. Explain how the subsystems will be analyzed for both inertia response and the response due to differential anchor movements.

GE Response

The differential displacements are obtained from the dynamic analysis of the buildings.

Displacements are applied to the anchors and restraints corresponding to the maximum differential displacements that could occur. The static analysis is made three times: once for one of the horizontal differential displacements, once for the other horizontal displacement, and once for the vertical.

The inertia (primary) and displacement (secondary) loads are dynamic in nature and their peak values are not expected to occur at the same time. Hence, the combination of the peak values of inertia load and the anchor displacement load is quite conservative. In addition, anchor movement effects are computed from static analyses in which the displacements are applied to produce the most conservative loads on the components. Therefore, the primary and secondary loads are combined by the SRSS method.

DCD Impact

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

NRC RAI 3.9-35

In DCD Tier 2, Section 3.7.2.3, for seismic analysis modeling, the amplified response spectra are generally specified at discrete building nodal points. No discussion is provided for the incorporation of any additional flexibility between these points and the pipe support (e.g., supplementary steel) in the piping analysis model. Provide a general discussion on the effects of this additional flexibility on the amplified response spectra, considering different varieties of pipe supports.

GE Response

Pipe supports are designed and qualified to satisfy stiffness values that are used in the pipe analysis. For struts and snubbers, the stiffness to consider is the combined stiffness of strut, snubber, pipe clamp and pipe support steel. For other type of supports, it is demonstrated that the support is dynamic rigid to preclude amplification. Also see response to RAI 3.12.32(2).

DCD Impact

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

NRC RAI 3.9-36

In DCD Tier 2, Sections 3.7.2.3 and 3.9.2.2.2, for the seismic analysis of piping systems, no discussion is provided for the situations when piping terminates at non-rigid equipment (e.g., tanks, pumps, or heat exchangers), and how the piping analytical model would consider the flexibility and mass effects of the equipment. Discuss how the flexibility and mass effects of the non-rigid equipment attached to the piping are incorporated in the analytical model.

GE Response

When piping terminates at non-rigid equipment (e.g., tanks, pumps, or heat exchangers), the six degree restraint stiffnesses at the attached point must be included in the piping analysis.

Normally, the tanks, pumps and the heat exchangers are anchored on floors. The thermal displacements at the pipe terminal ends must also be included in the analysis. When the dynamic displacements at the pipe terminal ends are sufficient small, the piping analysis is not significantly affected. In this case there is no need to consider the equipment mass in the piping analysis.

There are very few cases where tanks, pumps, or heat exchangers are not rigidly supported. If that is the case, the mass and the stiffness of the equipment is included in the piping model.

One example of a non-rigid component was the ABWR reactor internal pump (RIP). In this case, the piping model included the RIP to the RPV vessel wall attachment, and the RIP mass and the RPV nozzle stiffness were included in the analysis.

DCD Impact

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

NRC RAI 3.9-38

DCD Tier 2, Section 3.7.3.17, states that where small, Seismic Category II piping is directly attached to Seismic Category I piping, it can be decoupled from Seismic Category I piping. Provide the decoupling criteria.

GE Response

The Criteria for decoupling small branch lines from the main run of Seismic Category I piping is that small branch lines are decoupled from the main runs if the ratio of run to branch pipe moment of inertia is 25 to 1, or more.

DCD Impact

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

NRC RAI 3.9-39

DCD Tier 2, Section 3.7.2.13, states that the composite modal damping can be obtained either as stiffness-weighted or mass-weighted, and is limited to 20 percent. Explain how this 20 percent limit of damping was derived and how it can be justified.

GE Response

Twenty (20) percent limit of composite modal damping is a requirement of SRP 3.7.2.II.13.

DCD Impact

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

NRC RAI 3.9-40

In DCD Tier 2, Table 3.7-1 and Figure 3.7-36, a damping value of 20 percent is proposed for cable tray system (including supports) which are 50 percent to fully loaded. This is not consistent with the provisions of Regulatory Guide 1.61,Revision 0, October 1973. Revise the table and DCD Tier 2, Section 3.7.1.2, regarding the acceptable damping value for the cable tray system, including a detailed justification if the provisions of Regulatory Guide 1.61, Revision 0, October 1973, for damping are not met.

GE Response

Please see RAI 3.7-13 and RAI 3.7-13 Supplement 1 responses submitted under MFN 06-135 and MFN 06-135, Supplement 1. In DCD Tier 2 Revision 2, Figure 3.7-36 has been deleted and Table 3.7-1 has been revised.

DCD Impact

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

NRC RAI 3.9-44

In relation to DCD Tier 2, Section 3.9.2.2.2, "Qualification of Safety-Related Mechanical Equipment," for a dynamic loading event which involves both seismic and other reactor building vibration loads due to loss-of-coolant accident (LOCA) and safety relief valve (SRV) discharge, provide the combined required response spectra (RRS) for each major building floor, and for each major safety-related mechanical equipment. Discuss how these combined RRS are derived.

GE Response

Please see RAI 3.8-9 response submitted under MFN 06-298. The combined RRS for equipment qualification will be generated on a case-by-case basis in the detailed design phase using the SRSS method.

DCD Impact

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

NRC RAI 3.9-46

It is stated in DCD Tier 2, Section 3.9.2.3 that the major reactor internal components within the vessel are subjected to extensive testing, coupled with dynamic system analyses, to properly evaluate the resulting flow-induced vibration phenomena during normal reactor operation and from anticipated operational transients. However, a complete listing of the major components has not been provided. Provide this listing and identify each of the major reactor internal component within the vessel that would be subjected to flow induced vibration testing.

GE Response

DCD Tier 2 Appendix 3L.2 "Reactor Internal Components FIV Evaluation", subsection 1 "Evaluation Process - Part 1" identifies the reactor internal components for evaluation and potential FIV testing.

DCD/LTR Impact

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

NRC RAI 3.9-47

It is stated in DCD Tier 2, Section 3.9.2.3 that in general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Discuss GE's detailed analytical methodology to determine vibration forcing functions for obtaining operational flow transients and steady state conditions.

GE Response

The vibration forcing functions for operational flow transients and steady state conditions are determined by first postulating the source of the forcing function, such as forces due to flow turbulence, symmetric and asymmetric vortex shedding, pressure waves from steady state and transient operations. Based on these postulates, prior startup and other test data from similar or identical components are examined for the evidence of the existence of such forcing functions. Based on these examinations, the magnitudes of the forcing functions and/or response amplitudes are derived. These magnitudes are then used to calculate the expected ESBWR responses for each component of interest during steady state and transient conditions.

DCD Impact

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

NRC RAI 3.9-48

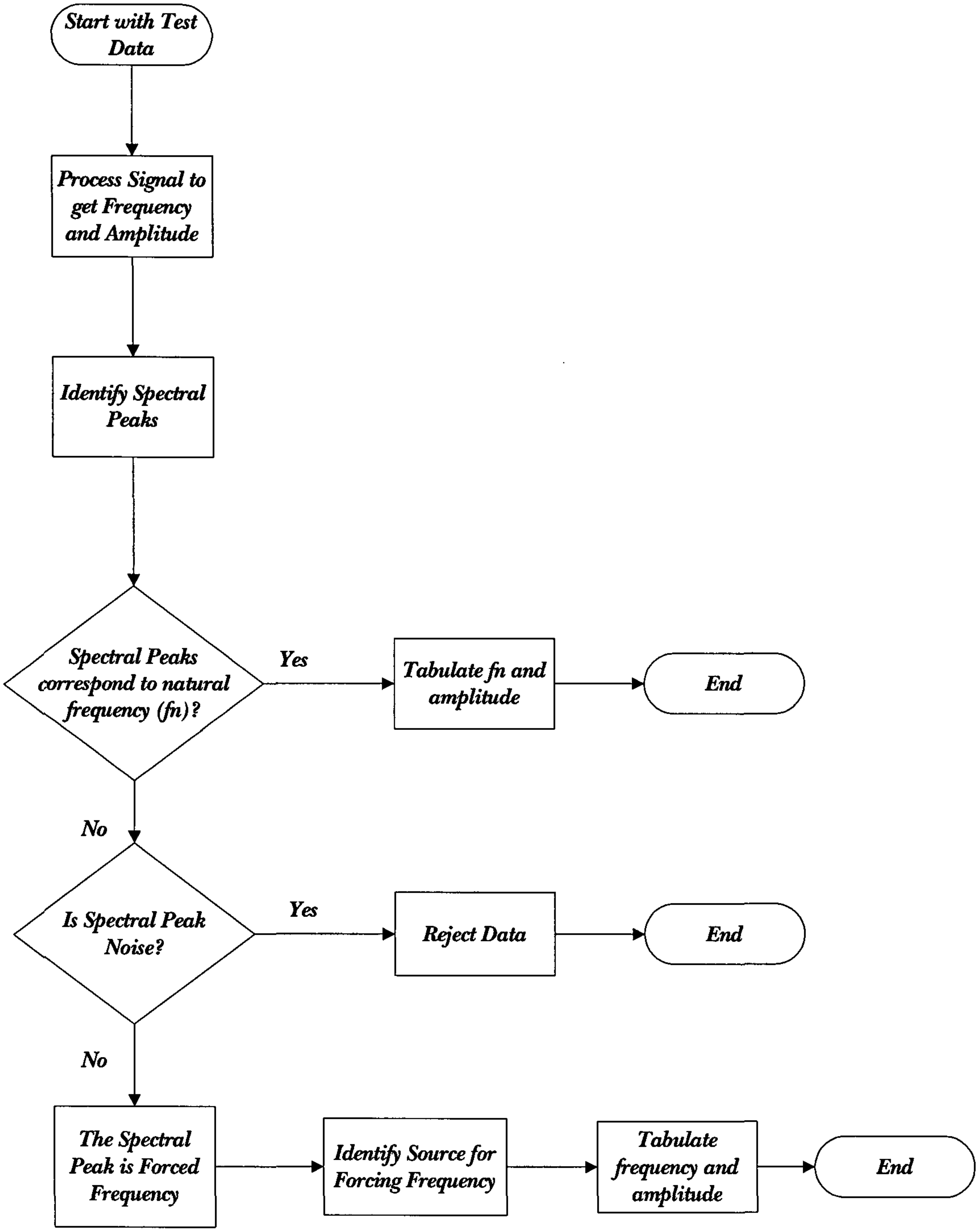
It is stated in DCD Tier 2, Section 3.9.2.3 that special analysis of the response signals measured from reactor internals of many similar designs is performed to obtain the parameters, which determine the amplitude and modal contributions in the vibration responses. Identify the specific parameters which are used to determine amplitude and modal contributions and explain with typical diagrams how these parameters are used in the special analysis.

GE Response

The test data from sensors (accelerometers, strain gages, and pressure sensors) installed on reactor internal components are first analyzed through signal processing equipment to determine the spectral characteristics of these signals. The spectral peak magnitudes and the frequencies at the spectral peaks are then determined. These spectral peak frequencies are then classified as natural frequencies or forced frequencies. If a spectral peak is classified as being from a natural frequency, its amplitude is then determined using a band-pass filter if deemed necessary. The resultant amplitude is then identified as the modal response at that frequency. This process is used for all frequencies of interest. Thus the modal amplitudes at all frequencies of interest are determined. If a spectral peak is identified are being from a forced frequency, the source (such as a vane passing frequency of a pump) is identified. Again, its magnitude is determined using a band-pass filter if deemed necessary.

The modal amplitudes and the forced response amplitudes are then used to calculate the expected ESBWR amplitudes for the same component. These ESBWR expected amplitudes are determined by calculating the expected changes in the forcing function magnitudes from the test component to the ESBWR component. For example, for flow turbulence excited components, the magnitudes are determined by ratioing with the flow velocity squared.

A flow chart of the above process is shown below.



DCD Impact

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

NRC RAI 3.9-49

It is stated in DCD Tier 2, Section 3.9.2.3 that response signals measured for reactor internals of many similar designs is performed to obtain the parameters, which determine the amplitude and modal contributions in the vibration responses. However, the specific plants which GE considers to be similar to the ESBWR design have not been specifically identified. Provide a listing of the plants which GE considers to have reactor internals similar to the ESBWR design and on what bases. Discuss the dissimilarities if any. Also discuss what impact they may have on the predicted results.

GE Response

The plants considered as being similar to the ESBWR depend on the component being investigated. For example, the incore monitor guide tube (ICMGT), and incore monitor housing, and CRGT in the ABWR, and all BWR5/6's are considered as being similar to the ESBWR. Except for shorter lengths due to a shorter core of the ESBWR, the designs for these components in these plants are essentially identical from a structural and FIV viewpoint. A shorter length will result in higher natural frequencies and lower responses for the ESBWR. For the shroud/separator structure, the ABWR design, except for the inclusion of the chimney in the ESBWR, is considered similar to the ESBWR. Inclusion of the chimney is expected to result in a different shroud/separator, chimney response for the ESBWR. Thus startup testing for this structure is planned.

The dissimilarities between the ABWR and the ESBWR are detailed in Table 2 of the Licensing Topical Report, NEDE-33259P, "ESBWR Reactor Internals Flow Induced Vibration Program – Part I", January, 2006.

DCD Impact

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

No changes to the subject LTR will be made in response to this RAI.

NRC RAI 3.9-50

It is the staff's understanding that the models used for the dynamic analysis of the reactor internals are similar to the analysis models used for Seismic Category I structures outlined in DCD Tier 2, Section 3.7.2. Discuss any differences that may exist between the analytical models being used in the dynamic analysis of major components and subassemblies, and the models used for Seismic Category 1 structures in Section 3.7.2 of the DCD Tier 2, document.

GE Response

The models for reactor internals, as well as Seismic Category I structures (DCD Tier 2, Subsection 3.7.2), are similar as far as the characterization of structural finite elements – mass, stiffness and damping – is concerned. However, their characterizations may take special forms more appropriate to the particular models. For example, damping of Seismic Category I structures may be better specified through composite material damping because of the widely different damping properties of structural materials in such structures, whereas the use of a simpler constant modal damping is realistic in the case of reactor internals. The nature of the forcing functions, which in the two cases is different, lends to mathematical simplicity in the case of seismic excitation, but is much more complex and random for pressure excitations in the ESBWR. Similarly, for Category I structures, a diagonal mass matrix is a standard representation of structural mass in the model, whereas for the analysis of RPV internals the inclusion of hydrodynamic masses coupling the degrees-of-freedom of internal components necessitate a non-diagonal representation of model mass matrix. The essential modeling procedures are, however, the same in both the cases.

DCD Impact

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

NRC RAI 3.9-51

It is stated in DCD Tier 2, Section 3.9.2.3 that data from previous plant vibration measurements are assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among boiling water reactors (BWRs) of differing size and design. Provide the extent of the variation in the response amplitudes,in BWRs of differing size and design for selected typical major reactor internals components.

GE Response

Since the shroud/separator structure is of special interest to the ESBWR, the variations in the measured shroud/separator responses during startup testing at full power for seven older reactors are provided below.

Plant Name	RPV ID (inches)	Shroud Displacement Amplitude (p-p mils)
Dresden 2	251	1.5
Dresden 3	251	1.5
Fukushima 1	188	0.5
Millstone	213	1.5
Monticello	205	1.0
Quad Cities 1	251	0.5
KKM	158	2.5

The mean value of these displacements is 1.29 and the standard deviation is 0.699.

DCD Impact

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

NRC RAI 3.9-52

It is stated in DCD Tier 2, Section 3.9.2.3 that parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates and structural parameters such as natural frequency and significant dimensions. Identify all the parameters which are expected to influence vibration response amplitudes among the reference plants. Also discuss the relative significance of each parameter.

GE Response

The following process parameters have the potential to impact component vibration amplitudes: power, re-circulation flow rates and velocities, feedwater flow rates and velocities, and steam mass flow rates and velocities. Plant transients are affected by MSIV and turbine stop valve (TSV) closure rates. The following structural parameters have the potential to impact component vibration amplitudes: Structural and fluid damping, structural natural frequencies, and mode shapes. Other parameters that may impact vibration amplitudes are: Frequency of the forcing function, amplitudes and spatial distribution of forcing functions.

In general, the vibration amplitudes are linearly related to the fluid mass and proportional to the square of fluid flow velocities. Transient response amplitudes are generally inversely proportional to the closure rates of MSIV's and TSV's. In general, the vibration amplitudes are inversely proportional to the frequency squared. Also, the lower natural modes generally have higher responses because the generalized forces are normally higher for the lower mode shapes. This is because generalized force is a measure of the energy input into the vibrating system by the applied force. The frequency of the forcing function becomes critical if it is near a natural frequency. This is because resonance or near resonance could occur. At or near resonance, the vibration amplitudes increase exponentially.

DCD Impact

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

NRC RAI 3.9-53

It is stated in DCD Tier 2, Section 3.9.2.3 that correlation functions of the variable parameters are developed such that, when multiplied by response amplitudes, they tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response. Discuss the development of the correlation functions for the major components and response modes with typical specific examples to show how multiplication by the response amplitude tends to minimize the statistical variability.

GE Response

Since all BWRs are geometrically similar, the BWRs that have been vibration tested represent very good models of other reactor internals to be tested. Therefore, a prediction based on prior test results can be made based on engineering evaluation of the parameters that are known to affect vibration response. For each internals component the following relationships are defined:

$$X_i = m_i [A_i]^a [B_i]^b [C_i]^c \ldots\ldots\ldots$$

where

X_i = Modified non-dimensional amplitudes at plant i

m_i = Measured amplitude at plant i

A_i, B_i, C_i, ... = value of the correlation parameter for plant i

a, b, c ...... = undetermined coefficients

The objective is to evaluate the correlation parameters (such as flow, power, stiffness, velocity, etc.) and appropriate coefficients that tend to cause all the X_i to be equal. Lacking this idealized solution, a set of correlation parameters and appropriate coefficients, which tend to reduce the dispersion of the X_i, can be used to reduce the statistical dispersion (m_i) among plants.

As an example, for the shroud, the correlation factors are the reciprocal of the shroud power density [(shroud diameter)^2 / power] and calculated shroud fundamental frequency (f_i). Using the following measured data for the older BWRs, the modified non-dimensional amplitudes are determined:

Plant Name	Shroud Displacement (p-p mils)	Calculated Frequency (Hz)	Reciprocal Power Density
Dresden 2	1.5	6.35	0.529
Dresden 3	1.5	6.35	0.529
Fukushima 1	0.5	7.20	0.518
Millstone	1.5	8.42	0.520
Monticello	1.0	6.55	0.512
Quad Cities 1	0.5	7.43	0.529
KKM	2.5	5.68	0.485

For coefficient a=3 and b=1, the statistical properties of the resulting X_i, the modified data have a

mean value = 1.16, and a

standard deviation = 0.267

The standard deviation of the unmodified data is 0.699. Thus the modified data is shown to have lower dispersion.

DCD Impact

No DCD changes will be made in response to this RAI

NRC RAI 3.9-54

It is stated in DCD Tier 2, Section 3.9.2.3 that the predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic modal analyses. Explain with typical analytical data that the predicted amplitude takes into account the degree of statistical variability.

GE Response

Please refer to the response to NRC RAI 3.9-53. Finite element models of reactor internal components that have been tested are made to determine the natural frequencies and mode shapes of these reactor internal components. The results are compared to the measured values in the reactor. Where deemed appropriate, the finite element models are refined so that the calculated values are closer to the measured values. For a component requiring new tests, their finite element models are developed following the methodology used for the components already tested. Using the finite element models thus developed, the responses are calculated. The calculated response values are taken to be the mean value of the response. The correlation functions, consisting of the correlation factors and correlation coefficients, are calculated as described in the response to NRC RAI 3.9-53. The calculated mean values and standard deviation are used, in conjunction with other variable data (e.g. fatigue strength) to assess the structural adequacy from an FIV viewpoint.

DCD Impact

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

NRC RAI 3.9-55

DCD Tier 2, Section 3.9.2.3 states that the dynamic loads caused by flow-induced vibration (FIV) from the feedwater jet impingement have no significant effect on the steam separator assembly. Analysis is performed to show that the impingement feedwater jet velocity is below the critical velocity. However no analytical methodology or quantitative data is provided. Provide quantitative analytical or test data to demonstrate that dynamic loads caused by FIV from the feedwater jet impingement have no significant impact on the steam separator assembly.

GE Response

The shroud head and steam separator assembly in GE BWR/6 plants is clamped in place by 28, 32 or 36 shroud head studs and nuts. Together, the stud, nut, shroud head bolt, locking collar and certain other components comprise the shroud head stud assembly. The shroud head bolt is an in-reactor tool used to torque and un-torque the stud. The bolt also provides a locking function to prevent rotation of the stud. Shroud head stud bolt wear has been found at all GE BWR/6 plants. Wear has been observed on the bolt splines, on the guide pins of the locking collar assembly and on the bolt shaft where it passes through the lower support ring. This is the only flow induced issue that has occurred in the steam separator assembly, and the problem was unique to BWR/6 plants due to its different design where the shroud head bolt was unloaded and free to vibrate during plant operation. Mockup testing by GE has confirmed that the wear was caused from vibration of the shroud head bolts as a result of feedwater flow impinging on the bolt shafts. The shroud head bolts in the ESBWR design are quite different from the BWR/6 design, and are the same fundamental design that all other BWRs have successfully operated with, and have not experienced any vibration problems. In this design, the components that are opposite the feedwater flow are fully loaded during plant operation.

DCD Impact

DCD Tier 2, Subsection 3.9.2.3 will be revised to delete the sentence, "Analysis is performed to show that the impingement feedwater jet velocity is below the critical velocity" as noted in the attached markup.

NRC RAI 3.9-57

No drawings of the ESBWR reactor vessel internals, core support structures,steam dryer and chimney showing locations where predicted stress and displacements would be calculated, has been provided in DCD Tier 2,Section 3.9.2. Provide drawings of the core support structures, the reactor vessel internals including the steam dryer, chimney and other individual steam dryer components. Identify the locations where stresses would be computed.

GE Response

Please see MFN 06-178 response to RAI 4.5-2, 4.5-18, 4.5-19 and 4.5-20. Also see Figures 3.9-4, 4.2-2, 4.2-3, 4.6-7, 5.3-3, 3L-1 and 3L-2.

FEM models of most reactor internals will be created. The FEM models will list those areas with highest stress. For other reactor internals with more straight forward geometry and load input, stress locations to be evaluated are either based on previous BWR experience or engineering judgment based on the component geometry.

DCD/LTR Impact

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

NRC RAI 3.9-59

FIV evaluation analyses are required for all components with significantly different features and loading conditions, per Regulatory Guide 1.20, Revision 2,May 1976, and SRP Section 3.9.2, Draft Revision 3, April 1996. Provide detailed descriptions of each of the components, their structural boundary conditions and finite element method modeling (including assumed damping), the flow conditions, the FIV load definitions, the modal characteristics, and results of the response analyses, including acceptance criteria.

GE Response

The following ESBWR Internals components will be instrumented and analytically evaluated for FIV since they are new components that are being used in the ESBWR design:

Shroud and Chimney

Due to the addition of a chimney, the ESBWR shroud, top guide, chimney, and chimney head/steam separator assembly are considered to be new or sufficiently different to require testing and analysis. The shroud/chimney/steam separator assembly is a freestanding structure; however, there are eight lateral restraints at the top of the chimney that transmit loads to the RPV. The 12 shroud support brackets also provide a load path from the shroud to the RPV. There are bolted connections at the shroud to top guide, top guide to chimney, and chimney to chimney head.

In order to determine the shroud vibration frequencies and mode shapes, an axisymmetric shell model, with each node having four degrees-of-freedom, is developed using the ANSYS computer code or an equivalent qualified program. The detailed shell model consists of the RPV, chimney, chimney support, and shroud, such that the hydrodynamic interaction effects between the components are accounted for.

This shell model is applicable only to the axisymmetric finite element analysis of the shroud and vessel. Responses calculated from this model, other than that of the shroud, shall not be construed as being representative of other reactor components.

The following assumptions are made in generating the axisymmetric shell model:

(1) Discrete components move in unison for guide tubes, steam separators, standpipes, and control rod drive housings and guide tubes.

(2) Masses are lumped at the nodal points. Rotational inertias of the masses are neglected.

(3) Stiffnesses of control rods, control rod drives, steam dryers, and incore housings are neglected.

(4) Top guide beam and core plate are assumed to have zero rotational stiffness.

(5) Masses of CRD housings below the vessel are lumped to the bottom head.

Equivalent shells are used to model the mass and stiffness characteristics of the guide tubes, steam separators, and standpipes such that they match the frequencies obtained from a horizontal beam model.

Diagonal hydrodynamic mass terms are selected such that the beam mode frequencies of the shell model agree with those from the beam model.

The RPV, chimney and shroud are modeled as thin shell elements. Discrete components such as guide tubes are modeled as equivalent thin shell elements. The shell element data are defined in terms of thickness, mass density, modulus of elasticity, and Poisson's ratio for the appropriate material and temperature.

The natural frequencies and mode shapes of the shroud shell model are given in terms of two parameters, termed "n" and "m". The "n" parameter refers to the number of circumferential waves, while the "m" parameter refers to the number of axial half-waves. Thus, for beam types of vibration, n=1.

The fluid velocity of the water in the annulus between the chimney and the RPV is approximately the same as that in the annulus between the shroud and the RPV for the ABWR; and therefore, the corresponding fluid induced forces are similar. In the ESBWR annulus between the shroud and the RPV, the fluid velocities are higher than those at the ABWR, and so are the fluid forces, because of a narrower annulus width.

The calculation of maximum FIV stresses in the shroud and the chimney requires, as a first step, the identification of modes that are excited by fluid forces. This information is obtained from strain gages and displacement transducers during testing. Using analytically determined mode shapes for the vibrating modes, the test data is then converted into maximum modal stress anywhere on the shroud and the chimney. The process is repeated for each vibration mode identified from the analysis of test data. The stresses for all vibrating modes are then appropriately combined to obtain total maximum stress. In the case when test data is not available, test data from the ABWR, suitably modified to account for differences in responses between the ABWR and ESBWR, is used.

The preceding analysis does not require the specification of damping since the effect of damping is implicit in test measurements. However, any supplementary analysis that may require the use of time histories of forcing functions, a 2 per cent damping will be used for FIV evaluation.

The GE acceptance criteria require that this maximum stress is below a threshold value of 68.9 MPa.

Standby Liquid Control Lines

In the ESBWR prototype plant reactor, there are two standby liquid control pipes that enter the reactor vessel and are routed to the shroud. To predict the vibration characteristic of the standby liquid control line, a dynamic finite element model of the entire line is developed. In the model the ends of the line are fixed anchor points since the lines are welded at the vessel nozzle and the shroud attachment points. The SLC pipe is modeled by beam elements with each node having six degrees-of-freedom. Pipe masses along with added fluid masses are lumped at nodes. The spacing of the nodes is determined by the expected stress gradient and the maximum frequency required to be predicted with accuracy.

The lower part of the SLC is subject to higher fluid forces than the upper part because the fluid velocity in the shroud-RPV annulus is higher than that in the chimney-RPV annulus.

The procedure for determining maximum stress is similar to that described above for the shroud/chimney FIV analysis; namely, identification of vibration modes from test data, analytical mode shape determination for thus identified modes, using test data and mode shape information to obtain maximum modal stress anywhere on the SLC lines, and combination of modal stresses to obtain the total maximum stress. Prior to the availability of test data, SLC piping responses are calculated by applying fluid forces based on ABWR measurements. Vortex shedding frequencies (lowest frequency=5.5 Hz.) are also calculated and compared to the calculated natural frequencies (lowest frequency=25.2 Hz.). As before, no damping is required in this analysis. However, a damping of 1 per cent will be used where required. The GE acceptance criteria require that this maximum stress be below a threshold value of 68.9 MPa.

<u>DCD Impact</u>

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

NRC RAI 3.9-60

It is stated in DCD Tier 2, Section 3.9.2.4 that vibration measurements are made during reactor startup at conditions up to 100 percent rated flow and power. Steady state and transient conditions of natural circulation flow operation are evaluated during the initial startup testing. However, the steady state and transient conditions of the natural circulation flow operation have not been provided. Provide a complete list of the steady state and transient conditions of the natural circulation flow operation which are evaluated.

GE Response

ESBWR is subjected to vibration testing during steady state at rated volumetric flow as well as transient conditions.

Transients outside of steady state include measurement during power ascension, also low, mid and high core power flow conditions. Vibrations are measured during AOOs such as turbine or generator trip, main steamline isolation, SRV actuation.

The internals vibration is measured during individual component or system startup testing where operation may result in significant vibrational excitation of reactor internals, such as Isolation Condenser testing. The duration of the startup testing at the various flow configurations shall ensure that each critical component vibration is within design limitations.

DCD/LTR Impact

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

NRC RAI 3.9-67

The discussion provided in DCD Tier 2, Section 3.9.2.4 does not specifically state that the startup test procedure will include specific hold points for interaction with the NRC staff. Verify that during the ESBWR startup tests the procedures will include specific hold points for interaction with the NRC staff. The activities to be accomplished during the power ascension should also be specified and the hold points should be of sufficient duration to accomplish those activities.

GE Response

Specific hold points for interaction with NRC staff will be included in the ESBWR startup procedures for FIV. DCD Tier 2, Subsection 3.9.9.1 commits to providing information on startup testing to the NRC at the time of COL application, and this commitment will included.

DCD Impact

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

NRC RAI 3.9-72

GE's FIV evaluation program for the reactor internals is incomplete and difficult to comprehend, because the FIV program information is spread over DCD Tier 2,Sections 3.9.5, 3.9.2, Appendix 3L and a supplemental report (MFN 06-012,NEDE-33259P). Also, the different documents are not cross referenced and, clearly, additional reports are planned. Provide a revised and comprehensive DCD on the FIV evaluation of reactor internals.

GE Response

The ESBWR Licensing Topical Report (LTR) for Vibration (NEDE-33259P) identified several components requiring additional analyses. These components are: Shroud/Chimney Assembly, Chimney Head/Steam Separator Assembly, and Standby Liquid Control (SLC) piping. The LTR will be updated upon completion of these additional analyses. Appendix 3L will be changed as necessary to be consistent with the LTR. In addition to the above components, the steam dryer and chimney partitions have their own separate programs.

DCD Impact

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

No changes to the subject LTR will be made in response to this RAI.

NRC RAI 3.9-73

In accordance with the guidance provided in Regulatory Guide 1.20, Revision 2,May 1976, and the SRP Section 3.9.2, Draft Revision 3, April 1996, the specifics of the instrumentation, the expected response, and the flow conditions for all components that will be instrumented during startup FIV testing, should be identified. Therefore the applicant is requested to provide the following additional information:

(a) identify each component which is being instrumented and explain why it is being instrumented

(b) provide the modal response characteristics and the specific locations and orientation of the sensors

(c) describe the sensors, including their sensitivities and frequency responses

(d) provide the expected response of the sensor for the flow conditions to be tested, as well as the test acceptance criteria for each sensor; and

(e) justify the use of the sensor and its placement.

GE Response

(a) The selection of the components to be instrumented is based on the following considerations:

- Is the component a significantly different or new design compared to earlier BWRs?
- Does the component have a history of FIV-related problems?
- Is the component subjected to significantly different or new flow conditions?

Based on these criteria, the following reactor internal components have been selected to be instrumented in the ESBWR startup FIV test program:

- Steam Dryer Bank Hoods and End Plates based on history of past FIV related problems (fatigue cracking between hood and endplate).
- Steam Dryer Skirt based on history of past FIV-related problems (fatigue cracking between skirt and drain channels).
- Steam Dryer Drain Channels based on history of FIV-related problems (fatigue cracking between skirt and drain channels).
- Steam Dryer Support Ring based on history of FIV-related problems (dryer rocking) and the resulting new design features for replacement dryer designs (e.g., strengthened weld joints, castings).
- Chimney partition assembly based on new design features (elongated chimney shell, partition assembly, chimney restraint), and potential new flow conditions.
- Chimney Head / Steam Separator assembly based on new design (flat head with beam reinforcement and elongated standpipes).

- Shroud /Chimney assembly based on new design features (discrete shroud support members and the chimney connection), potential new flow conditions and difficulty of repair in event of failure.
- Standby Liquid Control (SLC) internal piping based on new design.

(b) DCD Tier 2, Subsection 3.9.9.1 commits to providing information on startup testing to the NRC at the time of COL application. Subsection 3.9.9.1 will be modified at that time to provide the modal response characteristics and the specific locations and orientation of the sensors.

(c) Sensors to be used for ESBWR FIV test are:

- Strain gages
- Accelerometers
- Displacement Sensors – LVDT (Linear Variable Differential Transformer)
- Dynamic Pressure Sensors

All of the above sensors are designed for nuclear reactor environment. The selection and placement of the sensors will be based on past experience with other BWRs startup testing and analysis. The sensors will be pressure tested, and the ones that meet the requirements will be used for installation in to the reactor.

The strain gages are weldable type and will have a typical gage factor of 1.6, and they are capable of measuring up to 5000 micro-strain. These strain gages can be used for a frequency range between 0 to 2500 Hz. However, for ESBWR testing, the usable range will be limited to 2 Hz to 300 Hz bandwidth. The strain gage output sensitivity is typically set for 1 Volt to represent 100 micro-Strain.

The LVDTs will have typical measurement range of -200 to +200 mils with an overall frequency response from 2 Hz to 150 Hz. The transducer along with the signal conditioning would be field calibrated such that 1 Volt output to represent 10 mils displacement (typical).

The accelerometers are of piezoelectric type. The accelerometers have a typical sensitivity of 10 pC/G and have a range greater than 100 Gs. The usable measurement range for ESBWR testing will be limited to 10 Gs and will have overall frequency response of 3 Hz to 500 Hz. Accelerometer signals will also be double integrated for selected sensors to obtain displacement. The frequency response in displacement mode will be from 5 Hz to 500Hz. The typical overall output of the accelerometer together with remote charge converter and the amplifier would be set such that 1Volt equal to 2 G and 1Volt equal to 20 mils in displacement mode which are typical.

The pressure transducers are of piezoelectric type and will have typical sensitivity of 190 pC/bar for one type of transducer and 25 pC/bar for the less sensitive type. These dynamic pressure transducers are capable of measuring 20 bars or greater and have frequency response from 2 Hz to 1000 Hz. For ESBWR testing, the usable frequency bandwidth will be limited to 3 Hz to 500 Hz. The typical pressure range is expected to be less than 5 psi. The typical overall output of the pressure transducer together with remote charge converter and the amplifier would be set such that 1Volt equal to 1 psi.

(d) DCD Tier 2, Subsection 3.9.9.1 commits to providing information on startup testing to the NRC at the time of COL application. Subsection 3.9.9.1 will be modified at that time to provide the expected response of the sensor for the flow conditions to be tested, as well as the test acceptance criteria for each sensor.

(e) See answers to (a) and (b) above.

DCD Impact

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

NRC RAI 3.9-74

It is stated in DCD Tier 2, Section 3.9.2.4, that the information that will be provided to the NRC by the COL applicant on the Startup FIV Program, is provided in Section 3.9.7.1 of DCD Tier 2. However it is the staff's understanding that this information is really provided in section 3.9.9 - COL Information, Section 3.9.9.1 - Reactor Internals Vibration Analysis, Measurement and Inspection Program. The applicant is requested to clarify this discrepancy and amend DCD Tier 2 accordingly.

GE Response

The typographical error in the last paragraph of Subsection 3.9.2.4 will be corrected from 3.9.7.1 to 3.9.9.1.

DCD/LTR Impact

DCD Tier 2 Subsection 3.9.2.4 will be revised as noted in the attached markup.

NRC RAI 3.9-75

The use of the terms prototype and non-prototype in DCD Tier 2, Section 3.9.9.1 and GE Report MFN 06-012, NEDE-33259P are contradictory. Using Regulatory Guide 1.20, Revision 2, May 1976, revise DCD Tier 2, Section 3.9.9.1, including the information on startup testing that will be provided to the NRC.

GE Response

The term "prototype" in NEDE-33259P applies only to the shroud/chimney and SLC structures. The ESBWR as a whole is classified as Non-Prototype Category II. DCD Tier 2, Subsection 3.9.9.1 commits to providing information on startup testing to the NRC at the time of COL application. Subsection 3.9.9.1 will be modified at that time.

DCD Impact

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

No changes to the subject LTR will be made in response to this RAI.

NRC RAI 3.9-76

In accordance with Regulatory Guide 1.20, Revision 2, May 1976, and SRP 3.9.2, Draft Revision 3, April 1996, guidelines, differences between the valid prototype and the non-prototype reactors will have no significant effects on the vibratory response of any of the components. The applicant is requested to identify and describe the structures and flow conditions in the valid prototype which correspond to the ESBWR Feedwater Sparger and the Chimney-Head and Steam-Dryer Guide Rod, and provide additional evaluation and evidence to show that the differences, if any, have no significant effects on the vibratory response.

This information is considered pertinent in determining whether or not the ESBWR reactor internals can be classified as non-prototype Category II, in accordance with Regulatory Guide 1.20, Revision 2, May 1976, and SRP 3.9.2,Draft Revision 3, April 1996, guidelines.

GE Response

The ESBWR feedwater sparger, and the steam dryer guide rod are the same in design as the ESBWR prototype ABWR. BWR steam dryer guide rods, including those for the ABWR have had satisfactory operation for many decades and no FIV issues are anticipated. The feedwater spargers in older BWR's had encountered self-excited vibration problems due to leakage flow at the thermal sleeve. Subsequent to those occurrences, BWR feedwater spargers have been re-designed to eliminate or minimize leakage flow. Tests conducted on the re-designed spargers show negligible flow induced vibration response. Thus, even though the ESBWR feedwater flow is about 10% higher, no unacceptable vibration amplitudes are anticipated. There have not been any vibration issues with the re-designed feedwater spargers. The chimney head is a newly designed component. GE has completed additional analysis work on this component. The Licensing Topical Report NEDE-33259P will be revised to include the information on the analysis. This revision will be completed and submitted to the NRC by March 2007.

DCD Impact

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

NRC RAI 3.9-79

Comparing ESBWR's DCD Tier 2, Fig. 3.9-3 and ABWR's DCD Tier 2, Fig. 3.9-2,the character and distribution of the flow below the core can be expected to be different because of the lack of Jet Pumps and the presence of 12 separate Shroud Supports. Explain these flow differences and how they will not have a significant effect on the FIV response of these ESBWR safety related components. In particular, include a discussion of the potential effects of organized wake flows downstream of the Shroud Supports. This information is considered pertinent in determining whether or not the ABWR reactor internals design is a valid prototype of the ESBWR design in accordance with Regulatory Guide 1.20, Revision 2, May 1976, and SRP 3.9.2, Draft Revision 3,April 1996, guidelines.

GE Response

The flow within an ESBWR reactor vessel is driven by the hydraulic head within the reactor vessel. The absence of a recirculation pump to drive flow eliminates pressure pulses and turbulence from the pumps in prior BWR designs. In a forced circulation reactor with jet pumps, the high velocity jets cause additional disturbances in flow exiting from the jet pump diffuser. The flow exiting from the diffuser enters the lower plenum and excites the lower plenum components such as the CRGT and ICGT/Housing. On the other hand, the flow in the ESBWR, in the absence of pumping action, will have a much smoother lower velocity. Thus, the ESBWR flow entering the lower plenum has a lower velocity and flow disturbance lower than the flow in the ABWR. In addition to the above, the flow paths within the reactor vessel have better distribution and fewer flow disturbances due to the absence of jet pumps or reactor internal pumps, and have fewer changes in cross sectional area that cause flow variations. In the ESBWR, there are twelve shroud support brackets, each with a frontal area of 0.065 m^2. For the ABWR, there are 10 shroud support legs with a frontal area of 0.33 m^2 each. Thus wake turbulence in the ESBWR is much weaker. All the above factors, lower velocity and lower flow turbulence, combine to lower the FIV response of the lower plenum components.

DCD Impact

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

NRC RAI 3.9-80

There appears to be a discrepancy between GE Report MFN 06-012,NEDE-33259P, and GE Report MFN 05-116, DCD Tier 2, Appendix 3L, as to the need for additional evaluations of the In-Core Monitor Housings (ICMH) and the In-Core Monitor Guide Tubes (ICMGT) of the ESBWR, both of which are safety-related components. The Appendix 3L recommends modeling the components as a continuously connected structure to accurately predict the vibration characteristics. GE Report MFN 06-012, NEDE-33259P treats the components as individual tubes, both for determination of vibration characteristics as well as for fluid loading (a single tube in crossflow). Clarify the vibration characteristics and fluid loading on the ICMHs, the ICMGTs, and the Stabilizer Bar Network. Explain how the FIV response of the ESBWR's components will not be significantly different from those of the ABWR. This information is considered pertinent in determining whether or not the ESBWR reactor internals can be classified as non-prototype Category II, in accordance with Regulatory Guide 1.20, Revision 2, May 1976, and SRP 3.9.2,Draft Revision 3, April 1996, guidelines.

GE Response

To determine the natural frequencies and mode shapes, the incore monitor guide tubes and incore monitor housings are modeled by using beam elements interconnected by structural ties. On the other hand, individual cylinders of the incore monitor guide tubes and incore monitor housings are used for calculating the vortex shedding frequencies. Since the fundamental natural frequency of the incore guide tube forest is far removed from the vortex shedding frequency, the response excited by vortex shedding is small. Thus the dominant excitation is from flow turbulence. This is confirmed by the startup measurements made at the prototype ABWR plant. As pointed out in Section 5.6 of the LTR (NEDE-33259P), the ESBWR incore monitor guide tubes and housings are shorter than those in the ABWR. Thus the ESBWR structure has a higher fundamental frequency than that of the ABWR. The ESBWR velocity will be lower, and the vortex shedding frequency will be lower. Thus, the vortex shedding frequency will be even further removed from the natural frequency. Thus, no FIV issues are anticipated.

DCD Impact

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

No changes to the subject LTR will be made in response to this RAI.

NRC RAI 3.9-91

It is stated in DCD Tier 2, Section 3.9.2 that the knowledge gained from previous vibration tests has been used in the generation of the dynamic models for the ESBWR plant to predict vibration amplitudes, natural frequencies and mode shapes. Therefore, the applicant is requested to provide a comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals of the plant which GE considers to be similar to the ESBWR design, for possible verification of the mathematical model used in the analysis.

GE Response

The following comparison chart for ABWR demonstrates the adequacy of reactor internals models in predicting responses under operating conditions.

	First Mode Frequency (Hz)	
Component	**Analytical Prediction**	**Measured**
HPCF Coupling and Sparger	62.1	60.0
In Core Monitor Guide Tube	54 - 70	55 – 64.5
Control Rod Drive Guide Tubes and Housing	18.7 – 20.1	16 – 20

The shroud, which was modeled as a shell structure, shows a number of closely spaced modes with the lowest natural frequency of 6.8 Hz.. The test spectra of strain gages and displacement transducers on the shroud show dominant frequencies of 6 Hz, 9.5 Hz from the transducers, and 34 Hz and 41.5 Hz from the strain gages. These were identified as corresponding to analytically predicted frequencies of 6.8 Hz (n=2 harmonic, mode 1), 9.2 Hz (n=1 harmonic, mode 1), 34.1 Hz (n=1 harmonic, mode 3) and 40.6 Hz (n=2 harmonic, mode 3).

DCD Impact

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

NRC RAI 3.9-92

It is stated in DCD Tier 2, Section 3.9.2 that the knowledge gained from previous vibration tests has been used in the generation of the dynamic models for the ESBWR plant to predict vibration amplitudes, natural frequencies and mode shapes. Therefore, the applicant is requested to provide a comparison of the analytically obtained mode shapes with the shape of measured motion from the plant which GE considers to be similar to the ESBWR design, for possible identification of the modal combination or verification of a specific mode.

GE Response

An analysis of the vibration test data from sensors identifies the dominant vibration frequencies that correspond to either the component natural frequencies or forcing function frequencies. Unless each reactor internals component is extensively instrumented, it is not possible to determine mode shapes corresponding to dominant vibration modes exclusively from the test data. The recourse that is taken is to use analytical models that are validated by demonstrating agreement of predicted natural frequencies with those obtained from the test data. Once the test vibration natural frequency of a reactor internal is identified by the analytical model, the corresponding mode shape predicted by the analytical model is used to establish response characteristics of that internals component in that vibration mode. The relative magnitudes and phase relationships among the sensors on a particular component are used to help identify the correspondence between the analytic and test modes.

DCD Impact

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

NRC RAI 3.9-93

It is stated in DCD Tier 2, Section 3.9.2 that the knowledge gained from previous vibration tests has been used in the generation of the dynamic models for the ESBWR plant to predict vibration amplitudes, natural frequencies and mode shapes. Therefore, the applicant is requested to provide a comparison of the response amplitude time variation and the frequency content obtained from test and analysis conducted on the plant which GE considers to be similar to the ESBWR design, for possible verification of the postulated forcing function.

GE Response

As stated in GE response to RAI 3.9-92, the analytical model is validated by demonstrating agreement of predicted natural frequencies with those obtained from test. The analytical models cannot predict response amplitude time variation unless the forcing function time and spatial variation is known a priori. The quantitative assessment of this forcing function can only be made from the test data. The response time history and its spectral decomposition are obtained directly from the test sensor data at the sensor location. At other locations on the component, the analytically derived mode shapes enable the determination of responses, from those recorded at the sensor location.

DCD Impact

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

NRC RAI 3.9-94

It is stated in DCD Tier 2, Section 3.9.2 that the knowledge gained from previous vibration tests has been used in the generation of the dynamic models for the ESBWR plant to predict vibration amplitudes, natural frequencies and mode shapes. Therefore, the applicant is requested to provide a comparison of the maximum responses obtained from test and analysis conducted on the plant which GE considers to be similar to the ESBWR design, for possible verification of stress levels.

GE Response

As stated in GE response to RAI 3.9-92 and RAI 3.9-93, the analytical model is validated by demonstrating agreement of predicted natural frequencies with those obtained from test. The test sensor data from a reactor internals component, together with the analytically derived mode shapes, are employed to determine modal responses at locations other than the sensor location of the component. The modal responses are then appropriately combined to obtain maximum response anywhere in the component. The tests response is available only at the sensor location but provides a basis for the analytical determination of response elsewhere.

In an ABWR FIV study conducted in 1992, the analytical models for CRDGT/CRDH and ICMH were used to predict the maximum stress values. The first ABWR startup test data later confirmed the analytically predicted values to be realistic. The following table, comparing maximum stress responses obtained from analytical methods and those from startup test measurements, demonstrates the validity of analysis models, the methodology, and the reliability of results predicted by such models and methods

	Maximum Stress (Kg/mm^2)	Maximum Stress (Kg/mm^2)
Component	Analytical Results	Startup Test Results
CRDGT/CRDH	0.28	0.28
ICMH	1.2	1.0

DCD Impact

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

NRC RAI 3.9-96

Identify the differences in the tests that were conducted on the plant which GE considers to be similar to the ESBWR design and the ones that GE proposes to do on the first ESBWR plant. It is stated in DCD Tier 2, Section 3.9.2.4 that the first ESBWR plant will be instrumented for testing. However, it can be subjected to startup flow testing only to demonstrate that the flow induced vibrations similar to those expected during operation do not cause damage. The applicant is requested to explain why the testing for the first ESBWR plant is restricted only to those aspects that are perceived to demonstrate that the flow induced vibrations expected during operation do not cause damage. Identify the differences in the tests that were conducted on the plant which GE considers to be prototypical of the ESBWR reactor internals design and those tests which GE proposes to conduct on the reactor internal of the first ESBWR plant. It is the staff's understanding that GE contends that ESBWR reactor internals fall in the classification of Non-Prototype Category II. Therefore, the applicant is requested to discuss how its testing program is consistent with the vibration assessment program delineated in Regulatory Position C.2.2 of Regulatory Guide 1.20, Revision 2, May 1976, associated with the testing program for Non-Prototype Category II reactor internals.

GE Response

The ABWR was considered to be a prototype plant due to the introduction of reactor internal pumps and other new components. Also, higher power and higher core flows contributed to the ABWR being classified as a prototype plant. In accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.20, Rev. 2 for a prototype design, extensive analysis, testing and full inspection was conducted during the first plant startup. A total of 46 sensors of different types were used to obtain vibration data on 11 different reactor internals component structures. The ABWR components monitored during startup included the steam dryer, high pressure core flooder, control rod guide tube, the incore monitor guide tube and housing, the top guide, and the shroud. In addition, pressure sensors were installed at various locations. The pressure sensors are used to obtain data for potential diagnosis purposes.

For the ESBWR, extensive instrumentation of the chimney and standby liquid control lines, both non-prototypical components, is planned. Prior to the startup testing, extensive analyses of these two components are made to establish the acceptance criteria. The acceptance criteria are set such that the maximum stresses anywhere on the structure is less than 68.9 MPa. If the FIV response amplitudes are less than the acceptance criteria, damage to the component will not occur. Thus, the startup vibration program will ensure that these non-prototype components will not be subjected to unacceptable FIV stresses during operation.

DCD Impact

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

NRC RAI 3.9-97

DCD Tier 2, Table 1.9-22, identifies the 2004 edition of the ASME Code, Section III, as applicable to the design of components,component supports and core support structures. Confirm that the requirements of 10 CFR 50.55a(b) will be met without exception.

GE Response

Reference RAI 3.12-1 for response.

DCD/LTR Impact

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

NRC RAI 3.9-98

In DCD Tier 2 Table 3.9-2, footnote 12, provide justification for excluding seismic inertia loading in the calculation of ASME Code Sections NC/ND-3600 Equation (9), Service Levels A and B, and Equations (10) and (11).

GE Response

Seismic Inertia Loading is considered only for fatigue evaluation of ASME Code Class 1 components and core supports structures according to Table 3.9.2 for Service Level B.

For Class 2 and 3 Piping according to NC/ND-3600, seismic anchor motion loads producing secondary stresses in piping are limited through Equation (10b) in footnote 12 of Table 3.9.2.

For Class 2 and 3 Piping, Seismic SSE Inertia loads producing primary stresses in piping are not included for fatigue calculations in Service Level A and B but are included in Equation 9 for Service Level D. A SSE event producing a full range stress cycle is expected only 1 time in the plant life according to Table 3.9-1.

DCD Impact

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

<u>NRC RAI 3.9-99</u>

In DCD Tier 2, Section 3.9.3.1.1, provide the basis for the correlation between the plant conditions and the associated probability ranges. Also indicate the application of these probabilities to the events listed under Plant Events.

<u>GE Response</u>

As stated in the table contained in 3.9.3.1.1, ASME Code service level A events are "normal planned" events with a probability of 1.0. Correspondingly, events 1, 3, 4, 5, 6, 9, and 10 in Table 3.9-1 are plant events that are identified as ASME Code service level A events, and in all cases these are events that occur during every plant operating cycle. Next is ASME service level B, and as stated in the table contained in 3.9.3.1.1, these are events that have moderate probability between 10^{-2} and 1.0. For the ESBWR design life of 60 years this equates to events that are likely to occur between 1 to 60 events. Correspondingly, the plant events table 3.9-1 shows events 7, 8, 13, and 15 that have the number of events defined from 2 to 60 to occur in the 60 year plant life. For the Emergency (service level C) and Faulted (service level D) in the table contained in 3.9.3.1.1, the probability for these low and extremely low events is less than 10^{-2} (less than 1 in 100 years) to 10^{-6}. Therefore, any event in the plant events table 3.9-1 that is categorized as either service level C or D can not have more than 1 event in the 60 year design life of the plant (events 11, 12, 14, and 16). Each of these events are highly unlikely based on the event analysis provided for ESBWR. Therefore, these two tables are consistent in the basic methodology as to how to categorize events.

<u>DCD/LTR Impact</u>

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

NRC RAI 3.9-101

DCD Tier 2 Table 3.9-9 lists the acceptance criteria for Service Level A and B, Equations 12 and 13 as < 3.0 Sm. In DCD Tier 2 Section 3.9.3.3, confirm that the stresses in the main steam Class 1 piping meet these criteria.

GE Response

DCD Tier 2 Table 3.9-9 lists the Load Combinations and General Acceptance Criteria for Class 1 Piping Systems.

In addition, DCD Tier 2 Subsection 3.9.3.3 shows specific requirements for Class 1 Main Steam Lines in that Equation 12 and Equation 13 of NB-3600 must be less than 2.4 Sm. These requirements are according to Branch Technical Position MEB-3.1, which is required to avoid the postulation of rupture in High Energy Piping Lines.

DCD Impact

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

NRC RAI 3.9-102

Provide verification that DCD Tier 2, Sections 3.9.3 and 3.9.4 address all ESBWR pressure boundary safety-related components and component supports.

GE Response

Safety-related components within the pressure boundary and related component supports in Table 3.2-1 were compared with those listed in Subsections 3.9.3 and 3.9.4. It was not readily apparent that in-core instrumentation was included, so the RPV assembly definition in Subsection 3.9.3.2 will be changed to be consistent with subsection 3.9.1.4.

DCD/LTR Impact

DCD Tier 2 Subsections 3.9.1.4 and 3.9.3.2 will be revised as noted in the attached markup.

NRC RAI 3.9-104

In DCD Tier 2, Section 3.9.3.5, provide a description of Section 4.4 of the GE Environmental Qualification Program. Indicate if this program has been reviewed and approved by the NRC.

GE Response

NEDE-24326-1-P Licensing Topical Report (LTR) is referenced in ABWR Safety Evaluation Report (NUREG-1503), chapter 3.11.4 (page 3-95) which was approved by NRC.

DCD/LTR Impact

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

NRC RAI 3.9-105

In DCD Tier 2, Section 3.9.3.5, provide confirmation that the stresses in active valve bodies and pump casings loading conform with the requirements in Standard Review Plan (SRP) Section 3.10,Draft Revision 3, April 1996, for faulted conditions.

GE Response

It is confirmed that the stresses in active valve bodies conform to the requirements in SRP 3.10, Draft 3, April 1996. DCD Subsection 3.9.3.5 specifically identifies that the requirements of Section 3.10 are applicable, and this section identifies that testing and analysis is in compliance with SRP 3.10. The specific identification of the SRP Draft 3, April 1996 is found in Table 1.9-20 and reference 3.10-1. Since there are no safety related pumps in the ESBWR design, compliance with SRP 3.10 is not required for these components.

DCD/LTR Impact

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

NRC RAI 3.9-108

In DCD Tier 2, Section 3.9.3.6, provide verification that the design and installation of pressure relief devices is in accordance with the provisions in SRP 3.9.3, Draft Revision 2, April 1996, Section II.2.

GE Response

It is confirmed that the reference to SRP 3.9.3, which is contained in DCD Tier 2 Subsection 3.9.3, includes the provisions of SRP 3.9.3, Draft Revision 2, April 1996, Section II.2, within DCD Tier 2 Subsection 3.9.3.6 and that all the provisions are met in the ESBWR design. See Table 1.9-20 that confirms that this SRP draft revision is applicable for ESBWR design.

DCD Impact

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

NRC RAI 3.9-110

In DCD Tier 2, provide a list of the Design Reports documenting the qualification of the pressure relief devices. Confirm that the Design Reports meet the requirements stated in ASME Section III NCA 3550.

GE Response

Design Reports as required by ASME Section III NCA-3550 are provided as part of the delivery of completed N-stamped components. As such, these reports are not yet available, but in accordance with the NCA-3557 these reports are required to be available to the inspector at the plant site. Safety Relief valves provided for ESBWR will be in full compliance with the applicable ASME code requirements.

DCD Impact

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

<u>**NRC RAI 3.9-132**</u>

As discussed in DCD Tier 2, Appendix 3L (GE Report MFN 05-116),related to flow-induced vibration (FIV), GE indicated that many of the reactor internal components require additional analysis to demonstrate their design adequacy. Further, FIV evaluation analyses are required for all components with significantly different features and loading conditions from valid prototype reactor internals, per Regulatory Guide 1.20,Revision 2, May 1981, and SRP Section 3.9.5, Draft Revision 3,April 1996. GE is requested to provide detailed descriptions of the components, their boundary conditions, the load definitions, design criteria, bias errors and uncertainties, and the evaluation analyses for the ESBWR's Shroud/Chimney assembly, the Chimney Head/Steam Separator Assembly, the Standby Liquid Control lines, the Control Rod Guide Tubes and Housings, the In-Core Monitor Guide Tubes and Housings, the Chimney Partition, and the Steam Dryer.

<u>**GE Response**</u>

GE has completed additional analysis work on most of the components identified and will be revising the Licensing Topical Report NEDE-33259P to include the information requested. No analyses of the Control Road Guide Tubes and Housings, and the in-core Monitor Guide Tubes and Housings are deemed necessary. This revision will be completed and submitted to the NRC by March 2007.

<u>**DCD Impact**</u>

No changes to the subject LTR will be made in response to this RAI.

DCD Tier 2, Subsection 3.9.2.4 will be revised to delete startup testing of the CRGT/Housing and ICGT/Housing as noted in the attached markup.

NRC RAI 3.9-140

Section 3L.3 of DCD, Tier 2, Appendix 3L describes how GE has assessed the structural integrity of the Chimney Partition assembly using scale model testing, computational flow analyses, and finite element modeling and stress analysis. GE computed a maximum stress of 41 MPa using static analyses (based on their determination of a 2 Hz pressure fluctuation in the partition flow), which is less than the allowable 68.95 MPa established by ASME design codes. Details of the Chimney Partition evaluation analysis were not presented.

(a) *GE is requested to provide the flow conditions for which the twophase pressure measurements were made on the Chimney Partition and to provide the prototype conditions that they simulate, and describe the expected steam/water mixture flow rates and speeds through the Chimney Partitions. Also, provide the magnitude and frequency content of the associated loads. GE is requested to discuss how the loading conditions due to flow in the mixing chamber at the top of the chimney were included in the two-phase load definition on the partitions.*

(b) *GE is requested to explain how fluid loading (due to exterior water, and interior steam/water mixture) was considered in their finite element model (FEM), and the effects of the fluid loading on the model response at 2 Hz. Also, discuss the damping assumed in the chimney FEM, including the damping due to the fluid loading.*

(c) *GE is requested to describe the structural attachments and constraints of the Chimney Partitions and the Chimney, and to provide the justification for the modeling of the boundary conditions in the FEM analysis.*

GE Response

(a) The inlet flow conditions that were used in the test are shown on Table 1 that bound the actual flow conditions. The maximum load was measured in the 1/6 scale test and the value was 7.5 KPa (Peak-to-Peak/2) with 20% margin added, and the frequency was measured at 2 Hz.

Regarding the loading conditions due to the flow in the mixing chamber at the top of the chimney with respect to its effect on partitions, the test facility contained a tank that simulated the upper mixing chamber that was effective at collecting water as it existed the partitions. This test setup effectively modeled this interface, and simulates the pressure conditions that occur in the mixing chamber.

(b) In the FEM a pressure load of 7.5 KPa was applied uniformly on the plates, and the SRSS method was used for the sum of pressure loads between adjacent cells. See Figure 1 that shows the method used. This test was focused on the chimney partitions, and as such, the effect of exterior water was not considered.

Regarding the fluid loading, the Eigenvalues of the partitions are 53.8 Hz (276˚C) and 56.6 Hz (20˚C) without added mass, which is significantly higher than the 2 Hz dominant frequency of the fluid. Therefore, dynamic effects were neglected and a static analysis was performed, and no damping effects were considered.

(c) In the FEM model, the cells were modeled as integral elastic bodies, and the outermost ends of the partitions were assumed to have fixed ends. The detailed design of the chimney partition structure will include structural support components at the outmost ends of the partitions to provide rigidity.

DCD Impact

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

Table 1 Inlet conditions of the ESBWR chimney

Conditions		High Power	Average Power	Lower Power
Actual Velocity [m/s]	Gas	3.45	3.39	1.71
	Liquid	2.71	2.63	0.90
Void Fraction [-]		0.833	0.829	0.743
Superficial Velocity [m/s]	Gas	2.87	2.81	1.27
	Liquid	0.45	0.45	0.23

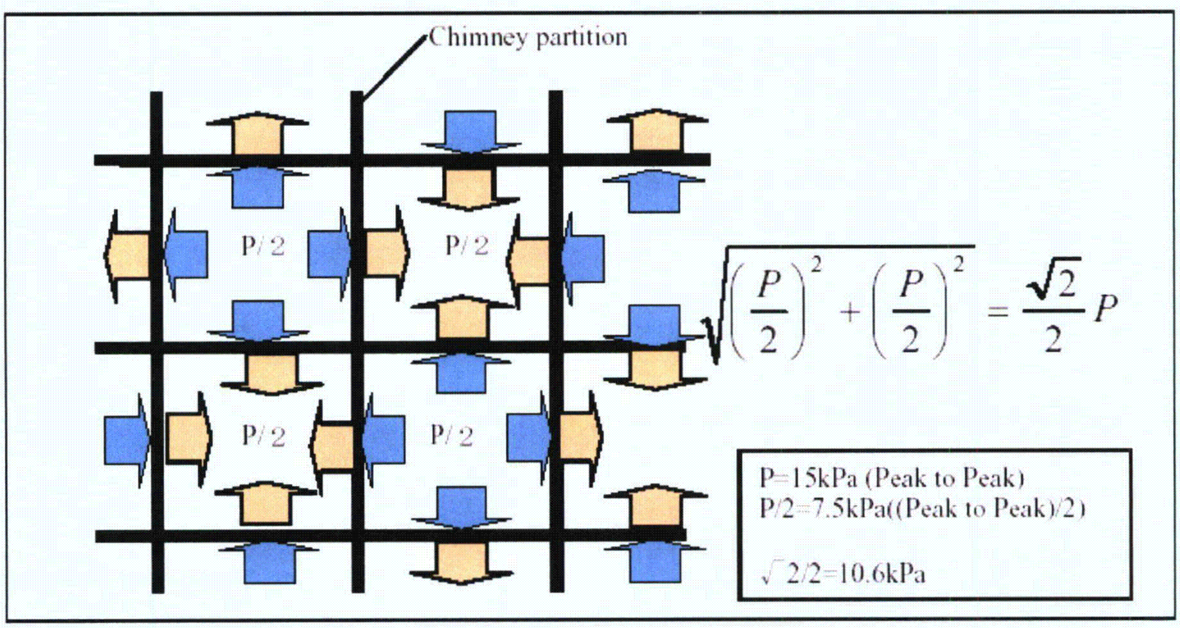


Figure 1 - Summation of Pressure Loads

NRC RAI 3.9-142

GE is requested to explain what fluctuating pressure loads are expected to emanate from the various nozzles in the reactor pressure vessel adjacent to the chimney. This explanation should include the Reactor Water Cleanup and Shutdown Cooling (RWCU/SDC) nozzle, the Isolation Condenser (IC) return nozzle, and the Gravity-Driven Cooling System (GDCS) nozzle near the chimney side walls as shown in Figure 2 on page 15 of the GE report (NEDE-33259P).

GE Response

Of the three systems that have nozzles and associated piping in the chimney region of the RPV, only the RWCU/SDC operates during normal plant operating conditions and has an external pump to drive flow. The other two systems are passive systems that do not operate during normal plant conditions and rely on hydraulic principles to create flow.

For the RWCU/SDC system, the RPV nozzle is used to remove water from the RPV during normal plant conditions. The flow rate in this mode is a maximum of 2% of the feedwater flow, and is provided by a pump with comparatively low capacity. The vane passing frequency from this pump will be similar to other pumps, but the amplitude will be very low. The BWR operating experience has been that only small sensing line components have been impacted by external pump vane passing frequencies.

The IC system is only operated when containment isolation occurs and heat removal from the reactor system is required. When this system is opened, steam flow drives each of the closed loops and flow enters the RPV from the IC return line nozzle. Plant operation with this system in operation will be very limited, and with the large mass of the chimney structure no flow induced vibration issues will occur.

For the GDCS lines, the only time these are placed in operation is during LOCA conditions when makeup water is required for the RPV. The flow from these nozzles is gravity driven from an elevated pool. The low associated flow rates and limited operating time, if such an event should ever occur, will not result in any vibration issues.

DCD Impact

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

NRC RAI 3.9-147

Since the natural circulation of the working fluid in the ESBWR is a new feature and only occurs when the fuel assemblies generate heat, GE is requested to justify that the flow velocities and their distribution over the reactor internals are verified for FIV analysis and testing, per SRP Section 3.9.2, Draft Revision 3, April 1996.

GE Response

The flow within an ESBWR reactor vessel is driven by the hydraulic head within the reactor vessel. This does not introduce any new adverse flow characteristics from the more traditional forced flow BWR reactors. In many respects, there are several positive aspects of natural circulation that would reduce the vibration amplitudes of the reactor internals. First, the absence of a pump to drive flow eliminates pressure pulses from the pump and other disturbances in flow that forced flow creates. Next the flow paths within the reactor vessel have better distribution and fewer flow disturbances due to the absence of jet pumps or reactor internal pumps, and have fewer changes in cross sectional area that cause flow variations. Also the flow velocity within the ESBWR core region is lower than forced circulation plants and the associated differential pressures across the components in the flow path are lower. Therefore, the vibration of reactor internal components due to flow disturbances within an ESBWR is expected to be lower.

DCD Impact

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

NRC RAI 3.9-150

Since there will be no preoperational FIV testing of the ESBWR because it operates in a natural recirculation mode (as noted in Section 3.9.2.4 of the DCD, Tier 2), GE is requested to discuss how the FEM's computed natural vibration modes (vibration predictions) of the reactor internal components will be correlated with test data, as specified in SRP Section 3.9.5, Draft Revision 3, April 1996, and SRP Section 3.9.2, Draft Revision 3, April 1996, Item 4.

GE Response

Prior to startup testing of reactor internal components, finite element models (FEM) of the reactor internal components to be tested are made. Using these FEMs, the natural frequencies and their corresponding mode shapes are computed by using computer programs for eigenvalue extraction. For each of these mode shapes, their locations of maximum displacement and maximum stress intensity are identified. For each accelerometer used during startup testing, and for each natural mode, the modal acceleration at the sensor location, the maximum modal acceleration, and the maximum modal stress intensity are identified. Based on this information, the vibration acceptance criteria for all accelerometers and all natural modes of interest are developed. A similar process is used for developing the acceptance criteria for strain gages. At the beginning of the startup test program for the first ESBWR, impact tests are performed to assess the acceptability of the information generated by the FEM. These impact tests are performed on instrumented components with an open reactor vessel under ambient conditions. The impact test results, including all natural frequencies and natural modes of interest, are compared to the FEM results using ambient condition fluid and structural properties. The results of the comparisons are used to refine the FEM if deemed appropriate.

DCD Impact

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

NRC RAI 3.9-151

GE is requested to describe the reactor internals vibration analysis, measurement and inspection information that the first COL applicant and the subsequent COL applicants need to provide, at the time of application, related to reactor vessel internals, including the core-support structures, beyond the information specified in Section 3.9.9.1 of DCD Tier 2. In addition, describe the plans for steam dryer instrumentation to confirm the stress analysis for ESBWRs to be constructed subsequent to the prototype.

GE Response

The program that GE intends to complete pertaining to FIV of reactor internal components is explained in Licensing Topical Report NEDE-33259P. This plan includes the completion of analysis for the remaining reactor internal components, and the details of the measurement and inspection program to be implemented at the startup of the first ESBWR plant. GE's plan is to complete this work in 2007 prior to submittal of the first COL submittal. Regarding the steam dryer FIV program, GE is planning to implement design features that will reduce the FIV susceptibility of the steam dryer, and commitments related to testing at subsequent ESBWR plants is not appropriate until all the evaluation work is complete.

DCD Impact

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

NRC RAI 3.9-153

Describe the method for the functional design and qualification of each safety-related pump for demonstrating the capacity of the pump to perform its intended safety function.

GE Response

ESBWR has no safety-related pumps.

DCD/LTR Impact

No DCD changes will be made in response to this RAI

3.8.4.1.6 Seismic Category I Cable Trays, Cable Tray Supports, Conduits, and Conduit Supports

Electrical cables are carried on continuous horizontal and vertical runs of steel trays or through steel conduits. The tray and conduit locations are based on the requirements of the electrical cable network. Trays or conduits are supported at intervals by supports made of hot or cold rolled steel sections. The supports are attached to walls, floor, and ceilings of structures as required by the arrangement. The type of support and spacing is determined by allowable tray or conduit spans which are governed by rigidity and stress. Bracing is provided where required. The loads, loading combinations, and allowable stresses are in accordance with applicable codes, standards, and regulations consistent with Tables 3.8-6 and 3.8-9. Design and location requirements for conduit and cable tray supports are also specified in Subsections 3.9.2 and 3.10.3.2.

3.8.4.1.7 Seismic Category I HVAC Ducts and HVAC Duct Supports

HVAC duct locations and elevations are based on the requirements of the HVAC system. HVAC ducts are made of steel sheet metal and are supported at intervals by supports made of hot or cold rolled steel sections. The supports are attached to walls, floor, and ceilings of structures as required by the arrangement. The type of support and spacing is determined by allowable duct spans that are governed by rigidity and stress. Bracing is provided where required. The loads, loading combinations, and allowable stresses are in accordance with applicable codes, standards, and regulations consistent with Tables 3.8-6 and 3.8-9. Design and location requirements for HVAC Ducts and HVAC Duct supports are also specified in Subsections 3.9.2, 9.4.1.3, 9.4.2.3 and 9.4.6.3.

3.8.4.2 Applicable Codes, Standards, and Specifications

3.8.4.2.1 Reactor Building

The major portion of the Reactor Building outside Containment structure is not subjected to the abnormal and severe accident conditions associated with a containment. Applicable documents for the RB design are shown in Table 3.8-9, except items 4, 11, 30 and 32.

3.8.4.2.2 Control Building

Applicable documents for the CB design are the same as the RB, which are listed in Table 3.8-9.

3.8.4.2.3 Fuel Building

Applicable documents for the FB design are same as the RB, which are listed in Table 3.8-9. Applicable documents for the spent fuel racks and associated structures are specified in Section 9.1.2.

3.8.4.2.4 Radwaste Building

Applicable codes, standards, specifications and regulations used in the design and construction of RW are items 1, 2, and 32 listed in Table 3.8-9.

- control rod drive outer tube; and
- bayonet fingers.

Only the bodies of the control rod guide tube, control rod drive housing and control rod drive outer tube are analyzed for energy absorption by inelastic deformation.

Inelastic analyses for the CRD housing attachment weld failure, together with the criteria used for evaluation, are consistent with the procedures described in Subsection 3.6.2 for the different components of a pipe whip restraint. Figure 3.9-1 shows the stress-strain curve used for the inelastic analysis.

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

This subsection presents the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, including those due to fluid flow and postulated seismic events discussed in SRP 3.9.2 draft R3. Structural requirements for Conduits and Cable Tray supports and HVAC Duct supports are specified in Subsections 3.8.4.1.6 and 3.8.4.1.7 respectively.

The plant meets the following requirements:

(1) GDC 1 as it relates to the testing and analysis of systems, components, and equipment with appropriate safety functions being performed to appropriate quality standards.

(2) GDC 2 as it relates to safety-related systems, components and equipment being designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena (SSE).

(3) GDC 4 as it relates to safety-related systems and components being appropriately protected against the dynamic effects of discharging fluids.

(4) GDC 14 as it relates to systems and components of the reactor coolant pressure boundary being designed to have an extremely low probability of rapidly propagating failure or of gross rupture.

(5) GDC 15 as it relates to the reactor coolant system being designed with sufficient margin to ensure that the reactor coolant pressure boundary is not breached during normal operating conditions, including anticipated operational occurrences.

3.9.2.1 Piping Vibration, Thermal Expansion and Dynamic Effects

The overall test program is divided into two phases: the preoperational test phase and the initial startup test phase. Piping vibration, thermal expansion and dynamic effects testing is performed during both of these phases as described in Chapter 14. Discussed below are the general requirements for this testing. It should be noted that because one goal of the dynamic effects testing is to verify the adequacy of the piping support system, such components are addressed in the subsections that follow. However, the more specific requirements for the design and testing of the piping support system are described in Subsection 3.9.3.7.

3.10.3.2 Other Electrical Equipment Supports

Supports for Battery Racks, Instrument Racks, Control Consoles, Cabinets, and Panels

Response spectra for floors where Seismic Category I equipment is located are supplied to each vendor. The vendor submits test data, operating experience, and/or calculations to verify that the equipment did not suffer any loss of function before, during, or after the specified dynamic disturbance. Analysis and/or testing procedures are in accordance with Subsection 3.10.2.

In essence, these supports are inseparable from their supported items and are qualified with the items or with dummy loads. During testing, the supports are fastened to the test table with fastening devices or methods used in the actual installation, thereby qualifying the total installation.

Cable Trays and Conduit Supports

Seismic Category I cable trays and conduit supports are designed by the response spectrum method. Analysis and dynamic load restraint measures are based on combined limiting values for static load, span length, and response to excitation at the natural frequency. Restraint against excessive lateral and longitudinal movement uses the structural capacity of the tray to determine the spacing of the fixed support points. Provisions for differential motion between buildings are made by breaks in the trays and flexible connections in the conduit.

The following loadings are used in the design and analysis of Seismic Category I cable tray and conduit supports.

- Loads
- Dead loads and live loads 112 kg/m (75 lbm/linear-ft) load used for 0.46-m (18-inch) and wider trays 75 kg/m (50 lbm/linear-ft) load used for 0.31-m (12-inch) and narrower trays.
- Dynamic loads - SSE loads plus appropriate RBV dynamic loads.
- Dynamic Analysis
- Regardless of cable tray function, all supports are designed to meet Seismic Category I requirements. Seismic and appropriate RBV dynamic loads are determined by dynamic analysis using appropriate response spectra.
- Floor Response Spectra — Floor response spectra used are those generated for the supporting floor. In case supports are attached to the walls or to two different locations, the upper bound envelope spectra are used. In many cases, to facilitate the design, several floor response spectra are combined by an upper bound envelope.

Structural requirements for Conduits and Cable Tray supports are also specified in Subsection 3.8.4.1.6.

Local Instrument Supports

For field-mounted Seismic Category I instruments, the following is applicable:

- The mounting structures for the instruments have a fundamental frequency above the excitation frequency of the RRS.

100% outside airflow. Any closed dampers in the fire area that are required to be open for purging smoke are reopened as part of the smoke purge mode. Area smoke detectors are provided in the CRHA and general areas of the Control Building.

Radiological Event Operation:

- When AC power is available, an outside air high radiation signal automatically starts the EFU fan and opens the normally closed outside air inlet dampers to the EFU. The signal also closes the normal outside inlet dampers to the AHU and closes the exhaust air dampers. All outside air is drawn through an EFU. Return air is routed to both the EFU and the AHU. During this mode of operation the EFU is monitored for low airflow and for radiation downstream of the EFU filters. Should either of these conditions be detected, the CRHA envelope is automatically isolated and EBAS is automatically actuated to supply breathing air and pressurization to the CRHA envelope. Also, due to the CRHA isolation, the MCR recirculation AHUs and CDUs automatically start; the EFU fans and the normal supply and return fans are automatically stopped and the restroom fans are manually stopped.
- When offsite and onsite AC power is lost (SBO), detection of an outside air high radiation condition initiates an automatic shutdown signal in the CRHAVS control system. The signal shuts down the supply AHU, the associated return/exhaust fan and sends a closure signal to the redundant CRHA isolation dampers. Due to the SBO event alone, the CRHA envelope is automatically isolated and EBAS is automatically started to supply breathing air and pressurization to the CRHA. The MCR recirculation AHUs/CDUs also start. The loss of all AC Power (SBO) causes the supply AHU, return/exhaust fan and restroom exhaust fan operation to stop. Note that the loss of AC power would cause the CRHA isolation dampers to revert to their fail-safe position (closed) and automatically actuate EBAS.

Toxic Gas Mode Operation:

- Upon detection of a toxic gas present at the outside air intake to the CRHAVS the outside air and exhaust dampers automatically close. The restroom exhaust dampers automatically close. Full return airflow is recirculated through the supply AHU. With the restroom exhaust flow path closed, the CRHAVS restroom exhaust fan is manually stopped.

9.4.1.3 Safety Evaluation

The CBVS is nonsafety-related except for the CRHA envelope, EBAS and associated instrumentation and controls, which are safety-related. The CRHA envelope includes structures, doors, components, ductwork between the CRHA envelope and the isolation dampers and the isolation dampers. The redundant isolation dampers fail closed upon a loss of control signal, power, or instrument air. The structural requirements for HVAC Ducts and HVAC Duct Supports are specified in Subsection 3.8.4.1.7

9.4.1.4 Testing and Inspection Requirements

Routine testing of components of the CBVS is conducted in accordance with routine power plant requirements for demonstrating system and component operability and integrity.

The FBFPVS AHUs are located in the Fuel Building HVAC Equipment Area.. The FBFPVS exhaust fans are located in the Reactor Building.

During high radiation conditions, the Fuel Building boundary isolation dampers close automatically and the supply AHU and exhaust fan shut down automatically in both subsystems.

System Operation

The FBVS operates during all normal, startup and shutdown modes of plant operation.

During normal operation, both the FBGAVS and FBFPVS subsystems are fully operable. Each subsystem operates with one supply AHU and one exhaust fan in service. The redundant AHU and fans are maintained on standby. In the event of low airflow in an exhaust duct, the standby exhaust fan starts. Simultaneously, due to a loss of negative pressure in the area, the AHU supply fan serving the area stops. The AHU supply fan restarts upon reestablishment of the required negative pressure. In the event of a fan failure, the failed fan automatically shuts down and the standby fan automatically starts.

On detection of high radiation, the Process Radiation Monitoring System provides a signal that trips the FBGAVS and FBFPVS subsystems. Each subsystem's supply AHU and exhaust fan shuts down and their associated dampers close. Exhaust air from either subsystem may be manually diverted to the Reactor Building HVAC purge exhaust filter unit. It is then exhausted to the plant vent stack by the Reactor Building HVAC purge exhaust filter unit exhaust fan. Normal ventilation for the area is resumed once the area is decontaminated or the source of radioactivity is removed.

If smoke is detected in a supply air duct, the affected subsystem is shut down. In the event of a fire, fire dampers close to isolate the fire area. Following a recovery from a fire, a subsystem exhaust fan removes smoke from the area by exhausting to the outside.

The FMCRD room AHU fan is started and stopped locally. A room thermostat modulates the chilled water valve in response to the room temperature.

An individual local thermostat controls each electric unit heater.

9.4.2.3 Safety Evaluation

The FBVS is not required to operate during a Station Blackout (SBO).

The Fuel Building boundary isolation dampers automatically close in the event of a fuel handling accident or other radiological accident. The safety-related isolation dampers fail closed upon a loss of control signal, power, or instrument air.

The FBVS components are designed as Seismic Category II, except for the safety-related isolation dampers and associated controls. The isolation dampers and associated controls are designed as Seismic Category I. The structural requirements for HVAC Ducts and HVAC Duct Supports are specified in Subsection 3.8.4.1.7

The FBVS does not have any safety-related functions, except for boundary isolation dampers closing in the event of radiological accidents. Redundant dampers and controls are provided so the refueling area is isolated upon demand even if a damper or control fails.

The RBVS components are designed as Seismic Category II, except for the safety-related building isolation dampers and associated controls. The building isolation dampers and associated controls are designed as Seismic Category I. The structural requirements for HVAC Ducts and HVAC Duct Supports are specified in Subsection 3.8.4.1.7

The RBVS does not perform any safety-related functions, except for boundary isolation dampers closing in the event of radiological events. Redundant dampers and controls are provided so the refueling area can be isolated even if one of the dampers or controls fail.

Rooms containing safety-related equipment have passive cooling features designed to limit the room temperature to the equipment's environmental qualification temperature.

9.4.6.4 Testing and Inspection Requirements

Routine testing of the RBVS is conducted in accordance with normal power plant requirements for demonstrating system and component operability. Periodic surveillance testing of safety-related building isolation dampers is carried out per IEEE-338.

The Reactor Building HVAC purge exhaust filter components are periodically tested in accordance with Regulatory Guide 1.140, Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants.

9.4.6.5 Instrumentation Requirements

The RBVS is operated from the MCR. A local run/stop control switch is provided for each fan for maintenance and testing purposes. The Reactor Building ventilation systems are manually controlled, except for certain automatic operations described below.

- Reactor Building boundary isolation dampers close on receipt a high radiation signal or on a loss of AC power
- For systems with redundant fans, the lead fan is selected manually. The standby fan automatically starts upon indication of low flow in the associated discharge duct.
- Fan operation is allowed only when the corresponding fan shutoff dampers (one upstream and one downstream) are open
- The CLAVS return/exhaust fan auto starts after the supply fan starts and the ventilated spaces are at a positive pressure
- Differential pressures between the ventilated spaces and the outside are transmitted to a pressure controller. The controller adjusts the CLAVS AHU supply fan's variable inlet vanes and airflow to maintain the ventilated at a positive pressure
- A temperature controller modulates the CLAVS outside, return and exhaust air dampers when outside air temperatures are below design supply air temperatures. Damper modulation provides a mixture of outside and return air at or below design supply air temperatures to the ventilated spaces
- The CONAVS supply fan auto starts after the exhaust fan starts and a negative pressure has been established in the ventilated spaces

plant. The predicted amplitude for each dominant response mode is stated in terms of a range, taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic modal analyses.

The dynamic modal analysis forms the basis for interpretation of the initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of ±68.95 MPa (±10,000 psi).

Vibratory loads are continuously applied during normal operation and the stresses are limited to ±68.95 MPa (±10,000 psi) to prevent fatigue failure. Prediction of vibration amplitudes, mode shapes, and frequencies of normal reactor operations are based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation.

The dynamic loads caused by flow-induced vibration from the feedwater jet impingement have no significant effect on the steam separator assembly. ~~Analysis is performed to show that the impingement feedwater jet velocity is below the critical velocity.~~ Also, it can be shown that the excitation frequency of the steam separator skirt is very different from the natural frequency of the skirt.

3.9.2.4 Initial Startup Flow-Induced Vibration Testing of Reactor Internals

Reactor internals vibration measurement and inspection program is conducted only during initial startup testing. This meets the guidelines of Regulatory Guide 1.20 with the exception of those requirements related to preoperational testing which cannot be performed for a natural circulation reactor.

Initial Startup Testing

Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Steady state and transient conditions of natural circulation flow operation are evaluated. The primary purpose of this test series is to verify the anticipated effect of single- and two-phase flow on the vibration response of internals.

Vibration sensor types may include strain gauges, displacement sensors (linear variable transformers), and accelerometers.

Accelerometers are provided with double integration signal conditioning to give a displacement output. Sensor locations include the following:

- Chimney and partitions, lateral displacements and accelerations;
- Chimney head, lateral displacements and accelerations;
- ~~Control rod drive housings, bending strain, lateral;~~
- ~~In-core housings and guide tubes, bending strain, lateral; and~~
- SLC internal piping, bending strain, lateral.

In all plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded and provision is made for selective on-line analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the

[NOTES IN GREEN HIGHLIGHT ABOVE SUBSECTION HEADINGS THAT CONTAIN CHANGES]

3.9.2.4 Initial Startup Flow-Induced Vibration Testing of Reactor Internals[RAI 3.9-74]

Reactor internals vibration measurement and inspection program is conducted only during initial startup testing. This meets the guidelines of Regulatory Guide 1.20 with the exception of those requirements related to preoperational testing which cannot be performed for a natural circulation reactor.

Initial Startup Testing

Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Steady state and transient conditions of natural circulation flow operation are evaluated. The primary purpose of this test series is to verify the anticipated effect of single- and two-phase flow on the vibration response of internals.

Vibration sensor types may include strain gauges, displacement sensors (linear variable transformers), and accelerometers.

Accelerometers are provided with double integration signal conditioning to give a displacement output. Sensor locations include the following:

- Chimney and partitions, lateral displacements and accelerations;
- Chimney head, lateral displacements and accelerations;
- Control rod drive housings, bending strain, lateral;
- In-core housings and guide tubes, bending strain, lateral; and
- SLC internal piping, bending strain, lateral.

In all plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded and provision is made for selective on-line analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer using frequency, phase, and amplitude information for the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes is then to be made on the basis of the analytically obtained normal mode that best approximates the observed mode.

The visual inspections conducted prior to, and remote inspections conducted following startup testing are for damage, excessive wear, or loose parts. At the completion of initial startup testing, remote inspections of major components are performed on a selected basis. The remote inspections cover the chimney, chimney head, core support structures, the peripheral control rod drive and incore housings. Access is provided to the reactor lower plenum for these inspections.

The analysis, design and/or equipment that are to be utilized for ESBWR comply with Regulatory Guide 1.20 as explained below.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program meets the requirements of Criterion 1, Quality Standards and Record, Appendix A to 10 CFR 50. This Regulatory Guide is applicable to the core support structures and other reactor internals.

Vibration testing of reactor internals is performed on all GE-BWR plants. Since the original issue of Regulatory Guide 1.20, test programs for compliance have been instituted for preoperational and startup testing. The first ESBWR plant is instrumented for testing. However, it can be subjected to startup flow testing only to demonstrate that flow-induced vibrations similar to those expected during operation do not cause damage. Subsequent plants, which have internals similar to those of the first plant, are also tested in compliance with the requirements of Regulatory Guide 1.20. GE is committed to confirm satisfactory vibration performance of internals in these plants through startup flow testing followed by inspection. Extensive vibration measurements in prototype plants together with satisfactory operating experience in all BWR plants have established the adequacy of reactor internal designs. GE continues these test programs for the generic plants to verify structural integrity and to establish the margin of safety.

Refer to Subsection 3.9.9~~7~~.1 for the information to be provided by the utility to the NRC on the reactor internals vibration testing program.

3.9.1.1 Considerations for the Evaluation of Faulted Condition[RAI 3.9-102]

All Seismic Category I equipment are evaluated for the faulted (Service Level D) loading conditions identified in Tables 3.9-1 and 3.9-2. In all cases, the calculated actual stresses are within the allowable Service Level D limits. The following subsections address the evaluation methods and stress limits used for the equipment and identify the major components evaluated for faulted conditions. Additional discussion of faulted analysis can be found in Subsections 3.9.2, 3.9.3 and 3.9.5.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits, such as clearance limits, are satisfied.

[ONLY RPV ASSEMBLY PARAGRAPH IS SHOWN]

Reactor Pressure Vessel Assembly

The reactor pressure vessel assembly includes: (1) the reactor pressure vessel boundary out to and including the nozzles and housings for FMCRD and in-core instrumentation; (2) vessel sliding support and (3) the shroud support ~~brackets~~. The design and analysis of these three parts complies with subsections NB, NF and NG, respectively, of the Code. For faulted conditions, the reactor vessel is evaluated using elastic analysis. For the sliding supports and shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly

3.9.3.2 Reactor Pressure Vessel Assembly [RAI 3.9-102]

The reactor vessel assembly includes: ~~consists of~~ (1) the reactor pressure vessel boundary out to and including the nozzles and housings for FMCRD and in-core instrumentation;~~,~~ (2) vessel sliding support, and (3) shroud support.

The reactor pressure vessel, vessel sliding support, and shroud support are designed and constructed in accordance with the Code. The shroud support consists of the shroud support brackets. The reactor pressure vessel assembly components are classified as ASME Class 1. Complete stress reports on these components are prepared in accordance with the Code requirements. NUREG-0619 is also considered for feedwater nozzle and other such RPV inlet nozzle designs.

The stress analysis is performed on the reactor pressure vessel, vessel sliding support, and shroud support for various plant operating conditions (including faulted conditions) by using the elastic methods, except as noted in Subsection 3.9.1.4. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are discussed in Subsection 3.9.5.

Table 3.9-1 [RAI 3.9-4, 3.9-11]

Plant Events

			ASME Code Service Limit[8]	No. of Cycles ~~Events~~[1]
A. Plant Operating Events [1, 9]				
1.		Boltup[1]	A	45
2.	a.	Hydrostatic Test (two test cycles for each boltup cycle)	Testing	90
	b.	Hydrostatic Test (shop and field)	Testing	3
3.		Startup (55.6°C/hr Heatup Rate)[2]	A	180
4.		Turbine Roll and Increase to Rated Power	A	180
5.		Daily and Weekly Reduction to 50% Power[1]	A	20,200
6.		Control Rod Pattern Change[1]	A	300
7.		Loss of Feedwater Heaters	B	60
8.		Scram:		
	a.	Turbine Generator Trip, Feedwater On, and Other Scrams	B	60
	b.	Loss of Feedwater Flow, MSIV Closure	B	60
9.		Reduction to 0% Power, Hot Standby, Shutdown (55.6°C/hr Cooldown Rate)[2]	A	172
10.		Refueling Shutdown and Unbolt[1]	A	45
11.		Scram:		
	a.	Reactor Overpressure with Delayed Scram (Anticipated Transient Without Scram, ATWS)	C	1[3]
	b.	Automatic Blowdown	C	1[3]
12.		Improper Plant Startup	C	1[3]
B. Dynamic Loading Events [1, 6, 9]				
13.		Safe Shutdown Earthquake (SSE) at Rated Power Operating Conditions	B[4]	~~2 Events~~ 20[5] ~~10 Cycles/ Event~~

Table 3.9-1 [RAI 3.9-4, 3.9-11]

Plant Events

		ASME Code Service Limit[8]	No. of Cycles ~~Events~~[1]
14.	Safe Shutdown Earthquake (SSE) at Rated Power Operating Conditions	D[7]	1 ~~Cycle~~[3]
15.	Safety/Relief Valve (SRV) Actuation (One) or single DPV actuation with depressurization (scram)	B	8
16.	Loss-of-Coolant Accident (LOCA):		
	Worst of small break LOCA (SBL), intermediate break LOCA (IBL), or large break LOCA (LBL)	D[7]	1[3]

Notes:

(1) Some events apply to reactor pressure vessel (RPV) only. The number of events/cycles applies to RPV as an example.

(2) Bulk average vessel coolant temperature change in any one-hour period.

(3) The annual encounter probability of a single event is $< 10^{-2}$ for a Level C event and $< 10^{-4}$ for a Level D event. Refer to Subsection 3.9.3.1.

(4) The effects of displacement-limited, seismic anchor motions (SAM) due to SSE shall be evaluated for safety-related ASME Code Class 1, 2, and 3 components and component supports. See Table 3.9-2 for stress limits to be used to evaluate the SAM effects.

(5) Use 20 peak SSE cycles for evaluation of ASME Class 1 components and core support structures for Service Level B fatigue analysis. Alternatively, an equivalent number of fractional SSE cycles may be used in accordance with Subsection 3.7.3.2.

(6) Table 3.9-2 shows the evaluation basis combination of these dynamic loadings.

(7) Appendix F or other appropriate requirements of the ASME Code are used to determine the Service Level D limits, as described in Subsection 3.9.1.4.

(8) These ASME Code Service Limits apply to ASME Code Class 1, 2 and 3 components, component supports and Class CS structures. Different limits apply to Class MC and CC containment vessels and components, as discussed in Section 3.8.

(9) Plant events applicable to 60 years.

Table 3.9-2 [3.6-22]

Load Combinations and Acceptance Criteria for Safety-Related, ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Structures

Plant Event	Service Loading Combination [1], [2], [3]	ASME Service Level [4]
1. Normal Operation (NO)	N	A
2. Plant/System Operating Transients (SOT)	(a) N + TSV	B
	(b) N + SRV[5]	B
3. NO + SSE	N + SSE	B[11], [12]
4. Infrequent Operating Transient (IOT), ATWS, DPV	(a) N[6] + SRV[5]	C[13]
	(b) N + DPV[7]	C[13]
5. SBL	N + SRV[8] + SBL~~[9]~~	C[13]
6. SBL or IBL + SSE	N + SBL (or IBL)~~[9]~~ + SSE + SRV[8]	D[13]
7. LBL + SSE	N + LBL~~[9]~~ + SSE	D[13]
8. NLF	N + SRV[5] + TSV[10]	D[13]

Notes:

(1) See Legend on the following pages for definition of terms. Refer to Table 3.9-1 for plant events and cycles information.

The service loading combination also applies to Seismic Category I Instrumentation and electrical equipment (refer to Section 3.10).

(2) For vessels, loads induced by the attached piping are included as identified in their design specification.

For piping systems, water (steam) hammer loads are included as identified in their design specification.

(3) The method of combination of the loads is in accordance with NUREG-0484, Revision 1.

(4) The service levels are as defined in appropriate subsection of ASME Section III, Division 1.

(5) The most limiting load combination case among SRV(1), SRV(2) and SRV (ALL). For main steam and branch piping evaluation, additional loads associated with relief line clearing and blowdown into the suppression pool are included.

(6) The reactor coolant pressure boundary is evaluated using in the load combination the maximum pressure expected to occur during ATWS.

(7) This applies only to the Main Steam and Isolation Condenser systems. The loads from this event are combined with loads associated with the pressure and temperature concurrent with the event.

(8) The most limiting load combination case among SRV(1), SRV(2) and SRV (ADS). See Note (5) for main steam and branch piping.

(9) ~~The piping systems that are qualified to the leak-before-break criteria of Subsection 3.6.3 are excluded from the pipe break events to be postulated for design against LOCA dynamic effects, viz., SBL, IBL and LBL.~~(deleted)

(10) This applies only to the main steamlines and components mounted on it. The low probability that the TSV closure and SRV loads can exist at the same time results in this combination being considered under service level D.

(11) Applies only to fatigue evaluation of ASME Code Class 1 components and core support structures. See Dynamic Loading Event No. 13, Table 3.9-1, and Note 5 of Table 3.9-1 for number of cycles.

(12) For ASME Code Class 2 and 3 piping the following changes and additions to ASME Code Section III Subsection NC-3600 and ND-3600 are necessary and shall be evaluated to meet the following stress limits:

$$S_{SAM} = i\,\frac{M_c}{Z} \leq 3.0\,S_h \quad (\leq 2.0S_y) \qquad \text{Eq. (12a)}$$

Where: S_{SAM} is the nominal value of seismic anchor motion stress

M_c is the combined moment range equal to the greater of (1) the resultant range of thermal and thermal anchor movements plus one-half the range of the SSE anchor motion, or (2) the resultant range of moment due to the full range of the SSE anchor motions alone.

i and Z are defined in ASME Code Subsections NC/ND-3600

SSE inertia and seismic anchor motion loads shall not be included in the calculation of ASME Code Subsections NC/ND-3600 Equation (9), Service Levels A and B and Equations (10) and (11).

(13) ASME Code Class 1, 2 and 3 Piping systems, which are essential for safe shutdown under the postulated events are designed to meet the requirements of NUREG-1367. Piping system dynamic moments can be calculated using an elastic response spectrum or time history analysis.

Load Definition Legend for Table 3.9-2	
Normal (N)	Normal and/or abnormal loads associated with the system operating conditions, including thermal loads, depending on acceptance criteria.
SOT	System Operational Transient (Subsection 3.9.3.1).
IOT	Infrequent Operational Transient (Subsection 3.9.3.1).
ATWS	Anticipated Transient Without Scram.
TSV	Turbine stop valve closure induced loads in the main steam piping and components integral to or mounted thereon.
RBV Loads	Dynamic loads in structures, systems and components because of reactor building vibration (RBV) induced by a dynamic event.
NLF	Non-LOCA Fault.
SSE	RBV loads induced by safe shutdown earthquake.

Load Definition Legend for Table 3.9-2	
SRV(1), SRV(2)	RBV loads induced by safety/relief valve (SRV) discharge of one or two adjacent valves, respectively.
SRV (ALL)	RBV loads induced by actuation of all safety/relief valves, which activate within milliseconds of each other (e.g., turbine trip operational transient).
SRV (ADS)	RBV loads induced by the actuation of safety/relief valves in Automatic Depressurization Subsystem operation, which actuate within milliseconds of each other during the postulated small or intermediate break LOCA, or SSE.
DPV	Depressurization Valve opening induced loads in the stub tubes and Main Steam system piping and pipe-mounted equipment.
LOCA	The loss-of-coolant accident associated with the postulated pipe failure of a high-energy reactor coolant line. The load effects are defined by LOCA1 through LOCA7. LOCA events are grouped in three categories, SBL, IBL or LBL, as defined here.
LOCA1	Pool swell (PS) drag/fallback loads on essential piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
LOCA2	Pool swell (PS) impact loads acting on essential piping and components located above the suppression pool water upper surface.
LOCA3	(a) Oscillating pressure induced loads on submerged essential piping and components during main vent clearing (VLC), condensation oscillations (COND), or chugging (CHUG), or (b) Jet impingement (JI) load on essential piping and components as a result of a postulated IBL or LBL event. Piping and components are defined essential, if they are required for shutdown of the reactor or to mitigate consequences of the postulated pipe failure without off-site power (refer to introduction to Subsection 3.6).
LOCA4	RBV load from main vent clearing (VLC).
LOCA5	RBV loads from condensation oscillations (COND).
LOCA6	RBV loads from chugging (CHUG).
LOCA7	Annulus pressurization (AP) loads due to a postulated line break in the annulus region between the RPV and shieldwall. Vessel depressurization loads on reactor internals (Subsection 3.9.2.4) and other loads due to reactor blowdown reaction and jet impingement and pipe whip restraint reaction from the broken pipe are included with the AP loads.
SBL	Loads induced by small break LOCA (Subsection 3.9.3.1); the loads are: LOCA3(a), LOCA4 and LOCA6. ~~See Note (9).~~
IBL	Loads induced by intermediate break LOCA (Subsection 3.9.3.1); the loads are: LOCA3(a) or LOCA3(b), LOCA4, LOCA5 and LOCA6. ~~See Note 9 of Table 3.9-2.~~
LBL	Loads induced by large break LOCA (Subsection 3.9.3.1); the loads are: LOCA1 through LOCA7. ~~See Note (9).~~