HITACHI

GE Hitachi Nuclear Energy

James C. Kinsey
Vice President, ESBWR Licensing

PO Box 780 M/C A-55
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057
jim.kinsey@ge.com

MFN 06-464, Supplement 7

Docket No. 52-010

February 4, 2008

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

Subject: **Response to Portion of NRC Request for Additional Information Letter Number 67 Related to ESBWR Design Certification Application - Dynamic Testing and Analysis of Systems, Components, and Equipment - RAI Numbers 3.9-49 S01, 3.9-53 S01, 3.9-59 S01, 3.9-72 S01, 3.9-73 S01 3.9-76 S01, 3.9-77 S01, 3.9-78, 3.9-79 S01 and 3.9-96 S01, and - Reactor Pressure Vessel Internals – RAI Numbers 3.9-132 S01 and 3.9-147 S01**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) originally transmitted via the Reference 1 letter and supplemented by an NRC request for clarification in References 2 and 3. The GEH response to RAI Numbers 3.9-49 S01, 3.9-53 S01, 3.9-59 S01, 3.9-72 S01, 3.9-73 S01, 3.9-76 S01, 3.9-77 S01, 3.9-78, 3.9-79 S01, 3.9-96 S01, 3.9-132 S01 and 3.9-147 S01 are addressed in Enclosure 1.

If you have any questions or require additional information, please contact me.

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

References:

1. MFN 06-378, Letter from U.S. Nuclear Regulatory Commission to Mr. David H. Hinds, Manager, ESBWR, General Electric Company, *Request For Additional Information Letter No. 67 Related To ESBWR Design Certification Application*, dated October 10, 2006.

2. E-Mail from C. Patel, U.S. Nuclear Regulatory Commission, to John Leatherman, GE, dated May 15, 2007 (ADAMS Accession Number ML071580015).

3. E-Mail from Chandu Patel, U.S. Nuclear Regulatory Commission, to John Leatherman, GE, dated May 10, 2007 (ADAMS Accession Number ML073050301).

Enclosure:

1. Response to Portion of NRC Request for Additional Information Letter Number 67 Related to ESBWR Design Certification Application - Dynamic Testing and Analysis of Systems, Components, and Equipment - RAI Numbers 3.9-49 S01, 3.9-53 S01, 3.9-59 S01, 3.9-72 S01, 3.9-73 S01 3.9-76 S01, 3.9-77 S01, 3.9-78, 3.9-79 S01 and 3.9-96 S01, and - Reactor Pressure Vessel Internals – RAI Numbers 3.9-132 S01 and 3.9-147 S01

cc:	AE Cubbage	USNRC (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRF	0000-0078-0388

Enclosure 1

MFN 06-464, Supplement 7

Response to Portion of NRC Request for

Additional Information Letter Number 67

Related to ESBWR Design Certification Application

Dynamic Testing and Analysis of Systems, Components, and Equipment

RAI Numbers 3.9-49 S01, 3.9-53 S01, 3.9-59 S01, 3.9-72 S01, 3.9-73 S01 3.9-76 S01, 3.9-77 S01, 3.9-78, 3.9-79 S01 and 3.9-96 S01

and

Reactor Pressure Vessel Internals

RAI Numbers 3.9-132 S01 and 3.9-147 S01

For historical purposes, the original text of RAIs 3.9-49, 3.9-53, 3.9-59, 3.9-72, 3.9-73, 3.9-76, 3.9-77, 3.9-79, 3.9-96, 3.9-132 and 3.9-147, and the GEH response is included, except for any attachments or DCD mark-ups.

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 3.9-49 S01

RAI 3.9-49 S01 Comment on response to RAI 3.9-49:

Based on its review of the referenced documents and the applicant's commitment to perform startup testing on components where necessary, the staff finds the applicant's response to perform testing where necessary is reasonable. However, the response and referenced documents are inadequate in identifying for flow induced vibrations (FIV) evaluation, and the similarities and dissimilarities between the components and flow conditions of ESBWR and other reactors.

The FIV response of a component depends on its structural characteristics (geometry, mass distribution, including added fluid mass, and boundary conditions) and the character of the pressures exerted on the component by the local flow field (as represented by pressure amplitudes, frequencies, spatial and time distributions and their correlations). In turn, the structural characteristics determine the modal characteristics (modal frequencies, mode shapes, modal masses, and modal damping) used in FIV evaluations. As the flow passes past the component and upstream flow obstructions and other components, the character of the flow (including the velocity vector field, the density, the viscosity, and the flow regimes) determines the pressures, FIV forcing functions, and FIV excitation mechanisms. All these variables should be discussed for each reactor component, when identifying similar components in other reactors that will be used for FIV evaluation. Using GE's examples of the incore monitor guide tube, the in-core monitor housings, and the control rod guide tube (CRGT), outstanding structural information includes a discussion of: the similarity of their boundary conditions, the similarity of their interconnections, of whether the components respond individually or in a group, and the similarity of the structural modal frequencies, mode shapes, modal masses, and modal damping. Outstanding fluid flow information includes a discussion of why the pressures exerted on these components by the natural convection flow in the ESBWR is expected to be similar to the near-field flow from the jet pumps in other reactors.

When FIV response results from other reactors are used to predict ESBWR component responses, complete justifications for the structural and flow similarities between the ESBWR and the other reactors for each ESBWR reactor component, should be provided. The structural justifications should include discussions of geometry, mass distribution, and boundary conditions, modal frequencies, mode shapes, modal masses, and modal damping. The fluid flow justifications should include discussions of pressure amplitudes, frequencies, spatial and time distributions and their correlations, the flow properties, the flow velocity vector fields, the flow regimes and the turbulent characteristics of the flow, and the potential FIV forcing functions and mechanisms.

As discussed above, the applicant should provide additional information to that provided in its response to RAI 3.9-49, comparing the components and flow conditions of ESBWR and other reactors so that reliable FIV evaluation can be made.

GEH Response

The stresses due to flow induced vibrations (FIV) of reactor internal structures are determined by their structural characteristics and the fluid forces acting on them. In the case of the ABWR, the characteristics of the structures are represented by appropriate finite elements in a finite element model (FEM). The FEM representations of the reactor internal structures are used to extract natural frequencies and mode shapes. These FEM-determined natural frequencies have been confirmed by startup test data in the first ABWR. Where appropriate, these natural frequencies and mode shapes can be used to determine the corresponding ESBWR natural frequencies and mode shapes.

In the case of ESBWR CRGT and ICGT, all the structural characteristics (geometry, solid and fluid mass distributions, material properties and boundary conditions) are, except for the overall length, identical to the ABWR. Because the ICGT's are joined together, they will vibrate as a unit in both the ABWR and ESBWR. ABWR test data show that this is indeed the case for the ABWR. Because of the large distances between the CRGT's, they will vibrate individually. This is again confirmed by ABWR startup test data. Since the structural natural frequencies are inversely proportional to the square of the overall length, the ESBWR frequencies can be calculated by using the ratio of length squared. Because the structural characteristics (geometry, solid and fluid mass distributions, and boundary conditions) are, except for the overall length, identical to the ABWR, the ESBWR mode shapes are the same except for the length parameter. Thus, these ESBWR natural frequencies and mode shapes can be used to determine the alternating stresses induced by FIV as detailed later.

The stresses due to flow-induced vibrations (FIV) of reactor internal structures are also determined by the fluid forces due to coolant flow. The dominant excitations are from vortex shedding and flow turbulence. The key parameter characterizing these excitations is the coolant flow velocity. Local coolant flow velocity and pressure characteristics are extremely difficult to determine analytically and are not considered to be sufficiently realistic. Rather than using analytically derived forces, the measured component responses of the ABWR are used to calculate the ESBWR ICGT and CRGT responses. The methodology is described below.

The method consists of dissecting the forced response equation

$$Y(t) = \sum_{i=1}^{\infty} \frac{\phi_i^T F(t, x)}{[(\omega^2 - \omega_i^2)^2 + 4\lambda_i^2 \omega^2 \omega_i^2]^{1/2}} \phi_i$$

for each term and determining the effect of length on the response. For the numerator, $\phi_i^T F(t, x)$, is the generalized force and it is directly proportional to the length, ϕ_i being the *ith* mode shape, t the time and x the spatial dimension. The damping in mode i is λ_i, the forcing frequency is ω and the *ith* natural frequency is ω_i. We can conservatively assume that $F(t, x)$ is the same for ABWR and ESBWR. In reality, $F(t, x)$ is higher for the ABWR because of its higher velocities than the ESBWR.

For the mode shape, $\phi_i(x)$, the displacement function can be differentiated twice with respect to x to arrive at the stress. For beams, the mode shape $\phi_i(x)$, is a linear combination of trigonometric and hyperbolic sines and cosines. In the case of pinned-pinned beam (CRGT) it is a trigonometric sine. In clamped-free case (ICGT/ICH), all the four functions show up in the representation for the mode shape. The argument of all these functions, for the *ith* mode, is $(A_i\, x / L)$ where L is the length of the beam, A_i is a constant, (A_i / L) is proportional to the square-root of the *ith* frequency and depends on stiffness and mass density of the beam. Upon differentiating twice with respect to x, the stress becomes inversely proportional to the length squared. This ratio is valid for all end conditions.

For the denominator, in the case of vortex shedding excitation, the actual frequencies for the ESBWR and ABWR can be used to get their ratio. In the case of turbulence excitation, the frequency ratios are conservatively assumed to be unchanged.

From the above equation, a calculation was performed to determine the ESBWR CRGT and ICGT responses from ABWR data. Table 4 from the ESBWR Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program." December 2007, which was transmitted to the NRC via letter Number MFN 07-635 dated December 7, 2007, is used as input to the calculation.

For turbulence-excited vibrations, the turbulence intensity is a function of the velocity squared. The turbulence frequency spectrum could be conservatively assumed to be the same for the ABWR and ESBWR.

Using the above method and inputs, the calculated stresses for the ESBWR are about 50% of the ABWR stresses for both the CRGT and ICGT/ICH structures. The maximum ESBWR zero to peak alternating stress intensities are 3.4 MPa for the ICGT, 14.6 MPa for the ICH and 5.1 MPa for the CRGT. These alternating stress intensities are well below the allowable value of 68.9 MPa.

For the shroud/separator structure, the ABWR design, is similar to the ESBWR, except for the inclusion of the chimney in the ESBWR. Inclusion of the chimney results in a different shroud/separator, chimney response for the ESBWR. Thus startup testing for this structure is planned.

The differences between the ABWR and the ESBWR are detailed in Table 2 of the Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program", December, 2007.

DCD Impact

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

Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program." December 2007, was transmitted to the NRC via letter Number MFN 07-635 dated December 7, 2007.

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:

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

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 3.9-53 S01

RAI 3.9-53 S01 Comment on response to RAI 3.9-53:

The staff finds the applicant's response to RAI 3.9-53 not completely acceptable because it has not fully justified the applicability of the data from other reactors to the ESBWR components. In particular, the statement, Since all BWRs are geometrically similar, the BWRs that have been vibration tested represent very good models of other reactor internals to be tested, is not necessarily true. Geometric similarity is only one consideration to determine if a component is a good model for ESBWR components.

Before using any data from other reactor components to develop correlations, the applicant should justify the similarity of each component using all parameters relevant to FIV excitation, as elaborated in RAI 3.9-49. Also, see applicant's response to NRC RAI 3.9-52 for a discussion of relevant parameters.

GEH Response

Please see the response to RAI 3.9-49 S01.

DCD 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.

DCD Impact

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

NRC RAI 3.9-59 S01

RAI 3.9-59 S01 Comment on response to RAI 3.9-59:

The applicant has provided, in response to RAI 3.9-59, descriptions of two components, their structural boundary conditions and FEM modeling (including assumed damping), the flow conditions, the FIV load definitions, the modal characteristics, and results of the response analyses, including acceptance criteria, as requested. The staff finds the applicant's response acceptable for the components reported, with three exceptions: (1) The applicant's response will be considered incomplete until the similarity for FIV evaluations are justified on a component-by-component basis. (2) The axisymmetric analysis of the freestanding shroud/chimney/steam separator structure does not allow evaluation of torsion modes. The applicant should provide justification that excitation of torsion modes is not significant and include a discussion of any torsion constraint between the chimney and the RPV at the lateral constraint as well as potential FIV excitation sources that could excite torsion modes, and (3) In RAI 3.9-59 the staff requested a response for all components with significantly different features and loading conditions, per RG 1.20 and SRP Section 3.9.2. The applicant should confirm whether the FEM evaluations provided in response to RAI 3.9-59 are the only components considered to have significantly different features and justify the exclusion of others.

GEH Response

The following ESBWR Internals components will be instrumented and analytically evaluated for FIV: shroud/chimney/steam separator assembly, standby liquid control lines, chimney internal partitions, and chimney head. (DCD Tier 2, Rev. 4, Section 3.9.2.4). These components have been determined to be sufficiently different from the earlier BWR's and the ABWR to warrant special attention.

The shroud/chimney/steam separator assembly is a freestanding structure with eight lateral restraints at the top of the chimney that provide translational and torsional restraints. The translation and torsional loads are transmitted to the RPV through the eight chimney support lugs. The shroud/chimney/steam separator assembly is essentially an axisymmetric structure and the flow is also axisymmetric. Hence, no significant torsional excitation is expected. Any minor torsional forces from the non-axisymmetric structural elements such as chimney internal partitions, and separator structural ties can be readily resisted by the torsional restraints. Since the ESBWR flow is more uniform than the ABWR, any torsional fluid forces would be even smaller than in ABWR. This, in addition to the torsional restraint at the top of the chimney, will result in an ESBWR torsional response that is less than the comparable ABWR response.

The components excluded from detailed FIV testing and analytical evaluation include the CRGT, and ICGT/ICH structures. Except for their shorter length in the ESBWR, these structures are identical to those at the ABWR. For these components, a detailed comparative evaluation against the ABWR show that the ESBWR FIV stresses are about 50% of those of the ABWR and much below the allowable value of 68.9 Mpa

(10,000 psi). The procedure for the detailed comparative evaluation is given in the answer to RAI 3.9-49 S01.

Also excluded from detailed FIV testing and analytical evaluation are the feedwater sparger, the steam separators, the top guide, and core plate. FIV considerations for the feedwater sparger are covered in RAI 3.9-76 S01, the steam separators are covered in RAI 3.9-56, the top guide is covered in RAI 3.9-77 S01, and the core plate is covered in RAI 3.9-78.

The only structures to be evaluated using testing and analytical evaluations are the shroud/chimney/steam separator assembly, the chimney partition, the chimney head, and the standby liquid control lines. All other structures in the ESBWR are considered to have characteristics sufficiently similar to the ABWR so that detailed FIV testing and analyses are unnecessary.

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 3.9-72 S01

RAI 3.9-72 S01 Comment on response to RAI 3.9-72:

The applicant's response is incomplete until analyses and testing for all internal components are provided, or component-by-component justifications are provided that show, as discussed earlier, a similar component has been tested in another reactor. Further, the applicant promises updates and revisions to documents after additional work is performed. The revised documents should be submitted for the staff review.

GEH Response

The response to RAI 3.9-59 S01 and Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program," December 2007, which was transmitted to the NRC via letter Number MFN 07-635 dated December 7, 2007, contains the information requested.

DCD Impact

No DCD changes 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 3.9-73 S01

RAI 3.9-73 S01 Comment on response to RAI 3.9-73:

A stated criterion for the selection of components to be tested is based on whether or not the component is subjected to significantly different or new flow conditions. In light of this and other criteria the applicant should justify why the components below the core (i.e., control rod guide tubes, in-core guide tubes and stabilizers, and non-pressure boundary portion of control rod housing and in-core housings) are not being instrumented for testing. These are critical safety related components and DCD information indicates that less turbulent convective flow passes by new core support structures before impinging on the core-control components. In particular, discuss potential FIV excitation mechanisms associated with the upstream core support structures. See also RAI 3.9-79.

GEH Response

Please see responses to RAI 3.9-79 S01, RAI 3.9-56 (submitted previously to the NRC in letter dated August 1, 2007, MFN 07-426), and RAI 3.9-49 S01.

DCD Impact

No DCD changes 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 redesigned 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 3.9-76 S01

RAI 3.9-76 S01 Comment on response to RAI 3.9-76:

The applicant's response is incomplete and needs clarification. The applicant has indicated that additional information will be provided for the chimney head. Further, clarification of the uncertainty associated with the terminology anticipated is required, as used in the phrase even though the ESBWR feedwater flow is about ten percent higher, no unacceptable vibration amplitudes are anticipated. We therefore have the following comment:

The applicant is requested to identify and describe the structures and flow conditions in the valid prototype that correspond to the chimney-head, and provide additional evaluation and evidence to show that the differences, if any, have no significant effects on the vibratory response. In addition, clarify the uncertainty in the use of the term anticipated in the phrase even though the ESBWR feedwater flow is about 10 percent higher, no unacceptable vibration amplitudes are anticipated. In particular, discuss whether past feedwater sparger leakage flow testing included the 10 percent higher flows of the ESBWR. If 10 percent higher flows were not achieved, provide the data/analysis that assures 10 percent higher flows will not lead to unacceptable vibration amplitudes. The occurrence of leakage flow instabilities is notoriously dependent on geometry of the leakage-flow paths, including alignment of the joined components, the vibration modes of the joined components, and especially small increases in flow rate. For further elaboration, see Leakage Flow-Induced Vibrations of Reactor Components, Shock and Vibration Digest, 15(9), 11-18 (1983). A conservative rule of thumb is to test leakage flow joints to 120 percent of operating flow rates.

GEH Response

The chimney head structure is unique to the ESBWR. As such, it is a newly designed structure requiring extensive analysis. A comprehensive finite element model has been used for the detailed evaluation. The analytical methodology and results are presented in Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program," December 2007, which was transmitted to the NRC via letter Number MFN 07-635 dated December 7, 2007.

As in the ABWR, the ESBWR feedwater (FW) sparger is of the "welded-in" design. In the welded-in design, flow leakage is eliminated by welding the thermal sleeve to the nozzle safe end. Since there is no leakage, there is no leakage-flow induced instabilities. Thus, even though the FW flow in the ESBWR is about 10% higher than that in the ABWR, there is no possibility of leakage flow induced instabilities.

DCD Impact

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

NRC RAI 3.9-77

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 describe the modifications made to the vibration analysis of the ABWR Top Guide Assembly used to predict the response of ESBWR's Guide. Demonstrate that the FIV response of ESBWR's Guide is not significantly modified by the structural differences with the ABWR's. In particular, discuss the modifications made to account for the differences in any cutout patterns in the Guides' plate, their diameters, and their attachments to the Shroud, and the Chimney or the Shroud Head. 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

This component has proven trouble free in past BWR designs, with various size cores, including the ABWR.

The design of the ESBWR Top Guide is made from a solid forging that is the same as the ABWR design in the arrangement and size of the cells. In addition, the overall thickness of the Top Guide is the same as the ABWR design. The ESBWR Top Guide does have a modestly larger overall diameter to accommodate the increased quantity of fuel assemblies and as a result, the ESBWR has a larger number of cells.

The Top Guide in both the ABWR and ESBWR is bolted into a larger structure. For the ABWR, the Top Guide is bolted to the shroud. For the ESBWR, the Top Guide is bolted between the shroud and chimney.

The flow across the Top Guide is limited to the by-pass flow between fuel assemblies. For the ESBWR the fluid velocities are lower than the ABWR, further reducing any potential for FIV.

DCD table 3L-4 identifies instrumentation that will be placed on the Top Guide to measure its lateral motion. This instrumentation will be the same as instrumentation placed on the Top Guide for the ABWR, as identified in DCD reference 3L-1.

DCD Impact

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

NRC RAI 3.9-77 S01

RAI 3.9-77 S01 Comment on response to RAI 3.9-77 (MFN 07- 207):

In its response to RAI 3.9-77, the applicant has not provided the engineering analysis or experimental evidence as requested in the RAI 3.9-77 and as required by Reg. Guide 1.20. In particular, an uninformative description of the ESBWR of the Guide Plate is given and it is only mentioned that the ABWR and ESBWR plates are connected to different components and the ESBWR plate has more cut outs and a larger diameter than the ABWR. Indeed, these are the issues that prompted the RAI. The ESBWR Guide plate supports in a cantilevered fashion a very long chimney on which the steam separators are attached. No long chimney (longer than either the shroud or separator components) exists in the ABWR between the Guide Plate and the Separators. Also, more of the Guide plate has been cutout in the ESBWR than in the ABWR, which may create greater stress concentration factors. Further, the fluid dynamic forces that will be transmitted to the Guide Plate will be different for the ESBWR: the fluid forces on the chimney do not exist in the ABWR and the steam separator unit is of a different design. The lateral motion of the guide plate is not of main interest; the dynamic stresses induced by dynamic deformation are of concern.

In accordance with Regulatory Guide 1.20, Revision 2, May 1976, and SRP-3.9.2, guidelines, show that the differences between the ABWR and the ESBWR Top Guide Plate will have no significant effects on the vibratory response. The applicant is requested to describe the modifications made to the analytical or experimental vibration analysis of the ABWR Top Guide Assembly used to predict the response of ESBWR's Top Guide Plate and demonstrate that the FIV response of ESBWR's top guide plate is not significantly modified by the structural and FIV loading differences with the ABWR's. If the information is not provided, then the ESBWR Guide Plate will be classified as a prototypic component, per Reg. Guide 1.20.

GEH Response

To calculate the FIV response of ESBWR Shroud/Chimney/Separator structure, measured pressure time histories in the ABWR RPV-Shroud annulus, were suitably scaled to define pressure time histories in the ESBWR RPV-Shroud/Chimney annulus. The scale factors were computed as the square of the ratio of ESBWR annulus fluid velocity to the corresponding value for the ABWR. Both ABWR shroud and ESBWR Shroud/Chimney structures were then analyzed under fluid forces resulting from the corresponding annulus pressure time histories to determine comparative responses of the Shroud/Chimney/Separator structure. During the prototype ABWR FIV test, the movement of the top guide was measured together with the shroud. The pressure time history was therefore further normalized such that the calculated ABWR response is equal to the measured ABWR response. The highest zero-to-peak stress intensity calculated on the basis of these measurements was 10.8 MPa.

The ESBWR top guide is made from a solid forging that is the same as the ABWR design in the arrangement and size of the cells. Since the calculated ESBWR lateral load at the top guide is 2.00 times higher than that of the ABWR, the highest zero-to-peak stress intensity is 21.6 MPa and is much below the allowable value of 68.9 MPa.

Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program," December 2007, was transmitted to the NRC via letter Number MFN 07-635 dated December 7, 2007 and includes the above information.

DCD Impact

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

NRC RAI 3.9-78

In GE Report MFN 06-012, NEDE-33259P, it is stated that the ESBWRs Core Plate requires no further evaluation for flow-induced vibrations (FIV), because it is similar to the ABWR's Core Plate which GE contends is a valid prototype of the ESBWR design. Show how the vibration analysis of the ABWR's Core Plate was modified to account for the structural differences with the ESBWR's Plate. Demonstrate that the ESBWR's FIV response is not significantly modified from the ABWR's. In particular, discuss the modifications made to account for any differences in the cutout patterns in the Core Plates, their diameters, and their attachments to the Shroud. The response should be in accordance with the guidelines of Regulatory Guide 1.20, Revision 2, May 1976, and SRP 3.9.2, Draft Revision 3, April 1996, which include that differences between the valid prototype and the non-prototype reactors will have no significant effects on the vibratory response of any of the components.

GEH Response

The ESBWR core plate assembly is of a similar design as that of the ABWR and BWR/6. The cutout patterns in the ESBWR, ABWR, and BWR/6 are identical. The core plate is a stiff structure that has no record of any FIV issues. The dynamic response analyses described in Response to RAI 3.9-77 S01 show that the ESBWR shroud acceleration at the core plate elevation is only 0.014g. The ABWR shroud acceleration at the core plate elevation is also at a negligible value of 0.009g. Since the core plate assembly is a very stiff structure, it will respond statically to the dominant shroud vibration frequency of 10.8 Hz. An ABWR core plate stress analysis shows that the maximum core plate stress is 3.7 MPa, well below the allowable value of 68.9 MPa. The ESBWR/ABWR core plate stress ratio is the same as the lateral load ratio of 2.55. Thus the maximum core plate stress is 9.4 MPa, well below the allowable value of 68.9 MPa. Thus, it can be concluded that the ESBWR core plate will have negligible stresses from FIV excitation of the shroud.

Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program," December 2007, was transmitted to the NRC via letter Number MFN 07-635 dated December 7, 2007 and includes the above information.

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 3.9-79 S01

RAI 3.9-79 S01 Comment on response to RAI 3.9-79:

The applicant's response is acceptable, with one exception, because turbulent flow differences and how they will not have a significant effect on the FIV response were presented. However, a discussion of the potential effects of organized wake flows downstream of the shroud supports was not given. The applicant needs to provide a satisfactory response to the staff concern discussed below:

According to the applicant's response, the ESBWR core flow has lower flow turbulence than the flow from past reactors that use jet pumps. But lower flow turbulence promotes the shedding of more organized shear layers from the smaller 12 shroud-support brackets upstream of the lower plenum components such as the CRGT and the ICGT/housing. The applicant is requested to discuss the potential effects of organized wake flows from the shroud supports shedding and impinging on downstream lower plenum components. In particular, include an assessment of the coincidence of the frequencies of organized wakes with the natural frequencies of the lower plenum components.

GEH Response

A calculation of the vortex shedding frequency from a shroud-support bracket shows that this frequency is less than 6 Hz. The components impacted by the wake flows from these brackets are the CRGT and the ICGT. The CRGT closest to a shroud-support bracket that would see the effect of this wake flow is 330 mm away, and the closest ICGT similarly placed with respect to the shroud-support bracket is 411 mm away.

These components are identical to the corresponding components in the ABWR except for their lengths, which are smaller in the case of the ESBWR. The shorter length in the ESBWR results in stiffer components and increased fundamental frequencies. The lowest estimated frequency for the CRGT, derived from its ABWR frequency value and adjusted for the effect of the length in the ESBWR, is 28.5 Hz and that for the ICGT, derived similarly, is 82.8 Hz.

The relatively low vortex shedding frequency and the much higher fundamental frequencies of components affected by the wake flows from the shroud-support brackets are sufficiently well separated to prevent any synchronization of the former with the latter. The possibility of a frequency "lock-in" resulting in resonance of CRGT or ICGT, therefore, does not exist.

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 3.9-96 S01

RAI 3.9-96 S01 Comment on response to RAI 3.9-96:

The applicant's response is unacceptable, because it only identifies the differences in the tests that were conducted on the ABWR, which the applicant considers to be prototypical of the ESBWR reactor internals design, and those tests that the applicant proposes to conduct on the reactor internal of the first ESBWR plant. The applicant did not explain why the testing for the first ESBWR plant is restricted only to those aspects that are perceived to demonstrate that the FIVs expected during operation do not cause damage. Further the applicant did not discuss how its testing program is consistent with the vibration assessment program delineated in Regulatory Position C.2.2 of RG 1.20, Revision 2, May 1976, associated with the testing program for Non- Prototype Category II reactor internals. The applicant should justify Non- Prototype Category II classification of the ESBWR on a component-by-component basis, as outlined in supplemental RAIs related to 3.9-49, 3.9-73, 3.9-76, 3.9-77, 3.9-78 and 3.9-79 and Confirmatory Item 3.9-72 related to this concern."

GEH Response

In LTR 33259P Revision 0, ESBWR Reactor Internals Flow Induced Vibration Program – Part 1, the ESBWR components requiring additional evaluations and tests for FIV, and components considered acceptable are delineated. The plant that is used for comparison purposes that is closest to the ESBWR configuration is the Advance boiling Water Reactor (ABWR). The first ABWR plant completed an FIV program that included analysis, testing and inspection as outlined in Regulatory Guide 1.20. Since the steam dryer and the chimney partition assemblies FIV programs were discussed in Reference 1 of the LTR, the LTR, Rev. 0, focused on the following components:

Chimney Head/Steam Separator Assembly

Shroud/Chimney Assembly

Top Guide

Core Plate

Standby Liquid Control (SLC) piping

Control Rod Drive Housings (CRDH)

Control Rod Guide Tubes (CRGT)

In-Core Monitor Guide Tubes (ICMGT)

In-Core Monitor Housings (ICMH)

The remaining reactor internals components that are not specifically identified in the reference 1, or in the LTR are basically proven by past trouble-free BWR experience, and have designs and flow conditions that are similar to prior operating BWR plants e.g. the feedwater spargers and guide rods (guides chimney head and steam dryer in place during installation).

An item by item discussion of why each component was considered to be prototypical and selected for further analysis and testing or why it was considered adequate without further detailed analysis or testing has been provided in Revision 1 of Licensing Topical Report NEDE-33259P, ESBWR Reactor Internals Flow Induced Vibration Program. The revised LTR contains detailed analytic methods used to determine the FIV response of each item requiring further evaluation, the results of the evaluation and comparison to allowable stresses. Where testing is determined to be required for a particular component, the revised LTR also includes the types and locations of sensors.

DCD Impact

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

NRC RAI 3.9-132

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.

GE Response

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.

DCD Impact

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 3.9-132 S01

RAI 3.9-132 S01 Comment on response to RAI 3.9-132 (MFN 06-464):

In its response to RAI 3.9-132, dated November 22, 2006, the applicant stated it will submit additional analysis work for most of the components identified in the RAI in the revision of Licensing Topical Report NEDE-33259P, scheduled for the release to the NRC in March 2007. The applicant has decided that no analyses of the Control Rod Guide Tubes and Housings, and the In-Core Monitor Guide Tubes and Housings are necessary. The Staff considers that the NRC RAI 3.9-132 is unresolved because the applicant has not yet submitted the additional analyses. The applicant should also explain why it would not perform any analyses of the Control Rod Guide Tubes and Housings, and the In-Core Monitor Guide Tubes and Housings.

GEH Response

Please see response to RAI 3.9-49 S01 for CRGT and ICGT and housings. Licensing Topical Report, NEDE-33259P Revision 1, "ESBWR Reactor Internals Flow Induced Vibration Program," December 2007, was transmitted to the NRC via letter Number MFN 07-635 dated December 7, 2007.

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 3.9-147

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 3.9-147 S01

RAI 3.9-147 S01 Comment on response to RAI 3.9-147 (MFN 06- 464):

Since the natural circulation of the working fluid in the ESBWR is a new feature and only occurs when the fuel assemblies generate heat, the staff requested the applicant in NRC RAI 3.9-147 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.

In its response to NRC RAI 3.9-147, dated November 22, 2006, the applicant explains how the working fluid flows in an ESBWR, and highlights positive aspects of the ESBWR design. The applicant states that the flow paths are cleaner in an ESBWR, with fewer flow disturbances. Also, the flow rates within the core region is slower than in forced circulation plant, leading to lower hydrodynamic excitation and resulting vibration.

The applicant's explanation of the benefits of ESBWR design regarding flow rates and patterns does not provide the information requested in the RAI, justify that the flow velocities and their distribution over the reactor internals are verified for FIV analysis and testing, per SRP 3.9.2. The applicant's response did not address the NRC RAI 3.9-147. Please respond to RAI 3.9-147 precisely.

GEH Response

Please refer to the RAI 3.9-49 S01 response. As stated in the RAI 3.9-49 S01 response, the dominant excitations of ESBWR reactor internal components are from vortex shedding and flow turbulence. The key parameter characterizing these excitations is the average coolant flow velocity. Hence, the average coolant flow velocity is used in the comparative evaluations. Local coolant flow velocities and pressure characteristics are extremely difficult to predict at analytically. Analytically derived pressure and forces are not likely to be sufficiently accurate. In place of purely analytically derived forces, the measured component responses during startup testing of the ABWR are used to comparatively calculate the ESBWR ICGT and CRGT responses as described in the response to RAI 3.9-49, S01. The use of ABWR test data to comparatively determine the ESBWR CRGT and ICGT/ICH responses is more reliable than theoretically derived responses.

DCD Impact

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