GE Energy

James C. Kinsey
Project Manager, ESBWR Licensing

PO Box 780 M/C J-70
Wilmington, NC 28402-0780
USA

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

MFN 07-202 Docket No. 52-010

April 4, 2007

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

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 43 Related to ESBWR Design Certification Application - ESBWR Probabilistic Risk Assessment - RAI Numbers 19.2-59 thru 19.2-61**

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

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

D068

Reference:

1. MFN 06-237, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 43 Related to ESBWR Design Certification Application*, July 18, 2006

Enclosures:

1. MFN 07-202, Response to Portion of NRC Request for Additional Information Letter No. 43 Related to ESBWR Design Certification Application - ESBWR Probabilistic Risk Assessment - RAI Numbers 19.2-59 thru 19.2-61

cc:	AE Cubbage	USNRC (with enclosures)
	George Stramback	GE/San Jose (with enclosures)
	Bob Brown	GE/Wilmington (with enclosures)
	EDRF Section	0000-0066-2057
		0000-0066-4780
		0000-0066-2059

Enclosure 1

MFN 07-202
Response to Portion of NRC Request for

Additional Information Letter No. 43

Related to ESBWR Design Certification Application

ESBWR Probabilistic Risk Assessment

Response to NRC RAIs 19.2-59, 60 and 61

NRC RAI 19.2-59

RAI Summary

Provide justification for the failure probability of 1E-3 given in Figure 8.3.3-4 (PRA 21.3.4.4-2)

Staff Assessment

In PRA, Revision 0, Section 21.3.4.4, GE stated that "...the detailed definition of a complete fragility that includes probability of failure as a function of load is rather superfluous." However, in Figure 8.3.3-4 of PRA Revision 0, the probability of failure of DCH, given RPV failure at HP and IC, steam line, SRVs intact, is presented as 1E-03. Provide justification for arriving at this failure probability, including a description of any analyses which may have been performed.

Audit Interest

Obtain an understanding of and assess the technical basis for the stated failure probability of 1 E-03.

GE Response

The stated reference has been superseded by material provided in NEDO-33201 Revision 1. The term "superfluous" was to convey a great margin to intersecting between DCH load and containment fragility. The numerical value of 10^{-3} was given to express a negligible probability of failure. In other words this came from the translation of ROAAM result of "physically unreasonable" to a numerical value. At the time it was "felt" that it was needed in subsequent PRA calculations. In the revision the need for such a translation was found to be superfluous, thus the result was left in the ROAAM "language" that DCH failure of the containment is "physically unreasonable". This change was made to Section 21 in revision 1, however Section 8 still showed the DCH_DAM node on the tree. NEDO-33201 Revision 2 eliminates all CET nodes demonstrated to be "physically unreasonable" from the calculations of LRF.

Changes to DCD or NEDO-33201

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

GE will make the calculation of LRF from Section 8 of the NEDO-33201, Rev 2 consistent with the phenomenology assessment in Section 21.

NRC RAI 19.2-60

Provide justification for the statement that liner must be strained to failure (typically 30% effective plastic strain) (PRA 21.4.4.1-1)

Staff Assessment

In PRA, Revision 0, Section 21.4.4.1, GE stated that "...to lose containment integrity, either the liner must be strained to failure (typically ~30% effective plastic strain)..." Provide justification for this statement, including the basis and possible test data to support the 30% failure effective plastic strain for liner materials.

Audit Interest

Ensure that a liner failure due to high strain has been correctly considered in the severe accident analyses. A strain limit of 30% does not appear to be appropriate, based on results reported in NUREG/CR-6906.

GE Response

This was a qualitative statement in introducing the subject. Complete definition of failure criteria is now provided in a supplement to PRA Chapter 21 (UCRL-TR-227386).

Further elaboration of failure criteria of steel under explosive loading is provided in the next 2 pages.

DCD Impact

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

THIS MATERIAL (SECTION 3.2) WAS ABSTRACTED FROM THE REFERENCE GIVEN BELOW AND INCLUDES THE REPORT FROM A ROAAM REVIEW PANEL EXPERT (SEE FIRST PARAGRAPH IN ATTACHED LETTER).

DOE/ID-I0541
VOLUME 2
June 1998

LOWER HEAD INTEGRITY UNDER IN-VESSEL STEAM EXPLOSION LOADS

T. G. Theofanous, W.W. Yuen, S. Angelini
J.J. Sienicki, K. Freeman, X. Chen, T. Salmassi

Advanced Reactor Severe Accident Program

Section: 3.2 Failure Criteria and Fragility

Failure criteria for ductile materials, as is the case here, have most commonly been based on plastic equivalent strains, with typically conservative "ball park" values in the 12 to 18% range (Theofanous et al. 1987). All experimental evidence, however, and theoretical interpretations indicate that failures are not obtained until much greater strains, say in the 50% to 100% range.

For example, Olive et al. (1979) explosively loaded cylindrical shells made of a variety of metals (including mild steel), and observed strains-at-failure in the 70% to 90% range. Pao and Gilat (1992) tested A533B steel specimens and found that failure did not occur until strains exceeded ~50%. Also with A533B steel, Shockey et al. (1980) found strains-at-failure in the 80 to 116% range. These results are particularly interesting because the A533B steel has only a slightly higher carbon content (0.19 vs. 0.16%), and a yield stress essentially equal to the as-tested value of the lower head material considered here.

The effect of stress anisotropy was evaluated by Korhonen (1987) on the basis of a metallurgical, plastic instability mechanism. Korhonen concluded that failure strain increases steadily from the isotropic value to nearly doubling as the principal stress ratio decreases to zero.

PAUL SHEWMON

[redacted] [redacted]

August 6, 1996

Re: Evaluation of report by T.G. Theofanous, et al., "Lower Head Integrity Under In-Vessel Explosion Loads", DOE/ID-10541

The analysis of head failure sets up a model of the lower head using ABAQUS (a well established finite element code) to relate the stress pulse from steam explosion to local strain. The vessel material (ferritic SA508 steel) will undergo large amounts of strain (elongations of 50 to 100%) before fracture occurs. Whether or not the vessel undergoes any plastic strain depends on the yield stress of the metal and the impulse from the steam.

For reason never explained or discussed, the authors chose 330 MPa for the yield stress of the vessel. They state that the conservative 'Code Allowable' is 345 MPa, and the actual value (found in a conventional tensile test) is 450 MPa. The choice of 330 MPa introduces a large conservatism (safety margin) since a best estimate should use 450 MPa.

The impulse applied to the steel in the lower head would have a rise time of a few milliseconds. When ferritic steels are loaded this quickly their yield stress is substantially greater than that observed in a normal tensile test. The authors quote references that show the yield stress at this strain rate is about 40% greater than that found in a tensile test. They take full credit for this strain-rate increment, which is justified and appropriate.

In summary, the analysis of head failure seems to be competently and conservatively done, and the conclusions drawn are appropriate. I have also looked at the discussion of loads and loading. I am less of a specialist in this area, but it also seems to be well done.

Though no mention of radiation effects is made in the report, the analysis should be made for the vessel at end of life (40 years?). The temperature of the head during the accident considered would be less than 212 F. This is beneath the RNDT for the beltline of some of the vessels now in service, i.e. such material might well behave in a brittle manner during an accident of the type considered here. I considered this, but feel such radiation effects are not germane in the case of the AP600 for at least two reasons:

1) The fast neutron and hard gamma flux in the lower head will be at least a couple of orders of magnitude less than that in the beltline region of the vessel, so radiation effects should be negligible.

2) The steel to be used in the AP600 vessel should be appreciably lower in the elements than have lead to radiation embrittlement (copper, and phosphorous) in the older vessels now of concern in plants in the U.S.A.

With this in mind, I believe there is every reason to believe that the material in the lower head would behave in a ductile manner and that the analysis given in the report is appropriate for (would apply to) a vessel in the AP600 after 40 years of service.

Original to L.W. Dietrich.

Paul Shewmon

NRC RAI 19.2-61

In PRA, Revision 1, Section 21.4.2, GE described that the key parameter for limiting ex-vessel steam explosion (EVE) loads is to limit the amount of sub-cooled water entering the lower dry well (LDW) before the melt resulting from the RPV lower head breach reaches to the LDW floor. GE described certain design changes, including preventing the GDCS overflow. However, GE's description of preventing the GDCS from overflowing emphasizes the system aspects, and does not discuss whether other natural phenomena such as earthquakes will induce the GDCS failure, leading to spilling water in the cavity. Provide the following discussions:

a) An explanation of whether or not the severe accident is initiated by a large earthquake, and if not, a discussion of postulating a condition under which a severe accident progress combined with an earthquake, and its effect on EVE.

b) A discussion of what the impact is of the pressure impulse due to EVE on the structural integrity of the hatch, given the equipment hatch on the pedestal being located at 2 m above the BiMAC cover plate, if the depth of sub-cooled water reaches above the equipment hatch.

c) Discuss any severe accident event sequence in which the initiating event is an earthquake greater than the safe shutdown earthquake (SSE) and failures or partial failures of GDCS pools, isolation condenser coolers, or suppression pool downcomers can occur prior to the accident progression to severe accident stage. If such an event sequence was considered, discuss the resulting containment pressure and temperature conditions, and show that the containment ultimate pressure capability is not challenged.

GE Response

a) An explanation of whether or not the severe accident is initiated by a large earthquake, and if not, a discussion of postulating a condition under which a severe accident progress combined with an earthquake, and its effect on EVE.

The risk due to a seismic event was evaluated using a seismic margins analysis (SMA) as described in Chapter 15 of NEDO-33201 Rev 1. This analysis concluded that the ESBWR plant and equipment are capable of withstanding an earthquake of a magnitude at least two times the safe shutdown earthquake (SSE) with a high confidence of low probability of failure. As discussed in the Containment Fragility Audit of February 5, 2007, the analysis is being revised to reflect a capability of 1.67 times the SSE. The DCD Tier 2 will be revised to reflect the new SMA.

The analysis in NEDO-33201 Rev 1 Section 21.4 did not assume the severe accident was initiated by a large earthquake. See response to c) below for a discussion of a severe accident caused by an earthquake more severe than evaluated in the SMA.

b) A discussion of what the impact is of the pressure impulse due to EVE on the structural integrity of the hatch, given the equipment hatch on the pedestal being located at 2 m above the BiMAC cover plate, if the depth of sub-cooled water reaches above the equipment hatch.

The assumption is made in NEDO-33201 Rev 1 Section 21.4 that if the lower drywell water level reaches 1.5 m, which is below the bottom of the equipment hatch, containment failure occurs with RPV failure. It should be noted that the PRA models assume that the floor of the lower drywell is at the BiMAC cover plate and does not credit the volume underneath the cover plate. Therefore, actual water level would be slightly lower than calculated. In addition, it would be expected that failure of the equipment hatch would occur at a threshold water level above the bottom of the equipment hatch.

c) Discuss any severe accident event sequence in which the initiating event is an earthquake greater than the safe shutdown earthquake (SSE) and failures or partial failures of GDCS pools, isolation condenser coolers, or suppression pool downcomers can occur prior to the accident progression to severe accident stage. If such an event sequence was considered, discuss the resulting containment pressure and temperature conditions, and show that the containment ultimate pressure capability is not challenged.

The SMA concluded that the ESBWR plant and equipment are capable of withstanding an earthquake of a magnitude at least two times the SSE with a high confidence of low probability of failure. As noted above, the SMA is being revised to reflect a capability of 1.67 times the SSE.

A severe accident event sequence in which the initiating event is an earthquake of a magnitude which causes the failures of the GDCS pools, isolation condenser coolers, or suppression pool downcomers would be assumed to cause containment failure and a release as described in NEDO-33201 Rev 1 Section 9.6.

DCD Impact

DCD Tier 2, Chapter 19 will be revised to show the new seismic margins analysis.