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 07-078

Docket No. 52-010

February 16, 2006

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

Subject: **Response to Portion of NRC Request for Additional Information Letters No. 40 and 59 Related to ESBWR Design Certification Application ESBWR - Probabilistic Risk Assessment - RAI Numbers 19.1-20 and 19.2-3**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the referenced letters.

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

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Reference:

1. MFN 06-222, Letter from NRC to David Hinds, *Request for Additional Information Letter No. 40 Related to ESBWR Design Certification*, dated July 5, 2006
2. MFN 06-329, Letter from NRC to David Hinds, *Request for Additional Information Letter No. 59 Related to ESBWR Design Certification*, dated September 13, 2006

Enclosures:

1. MFN 07-048, Response to Portion of NRC Request for Additional Information Letters No. 40 and 59 Related to ESBWR Design Certification Application ESBWR - Probabilistic Risk Assessment - RAI Numbers 19.1-20 and 19.2-3

cc: AE Cubbage USNRC (with enclosures)
David Hinds GE/Wilmington (with enclosures)
eDRF 0000-0060-5641
0000-0059-2978

**Enclosure 1**

**MFN 07-078**

**Response to Portion of NRC Request for**

**Additional Information Letters No. 40 and 59**

**Related to ESBWR Design Certification Application**

**ESBWR Probabilistic Risk Assessment**

**RAI Numbers 19.1-20 and 19.2-3**

**NRC RAI 19.1-20**

*Provide definitions, tables and simplified diagrams in Section 4.5. Please define acronyms, such as SSLC, RTIF and DPS, when they are first used in the text in Section 4.5 of NEDO-33201, and avoid using designators, such as B32, instead of the system name. Section 4.5 should be clarified by providing definitions, tables and simplified diagrams, to facilitate staff review of how the instrumentation and control (I&C) system was modeled in the PRA.*

**GE Response**

The definitions of the acronyms are Safety System Logic and Control (SSLC), Reactor Trip and Isolation Function (RTIF), and Diverse Protection System (DPS). The Distributed Control and Information System (DCIS) block diagrams, and the use of system names instead of designators such as B32, will be included in NEDO-33201 Revision 2.

**DCD/LTR Impact**

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

NEDO-33201 will be revised as described above in Revision 2.

**NRC RAI 19.2-3**

*A generalized description of severe accident progression is not provided in the PRA, and is needed to assist in evaluating the approach used in the PRA for describing and quantifying potential core damage sequences. In this regard, provide a narrative description, with quantitative estimates of times, temperatures and pressures, of the complete progression (in-vessel and ex-vessel) of a representative severe accident sequence in ESBWR. Compare and contrast the ESBWR accident response with that for a comparable accident sequence in the advanced boiling water reactor (ABWR).*

**GE Response:**

**Sequences of Each Accident Class**

As discussed in Section 8 of NEDO-33201 Revision 1, severe accidents were grouped in five categories in the PRA Level 1 analysis. The Level 1 analysis results were reviewed to identify sequences, which were dominant contributors to the core damage frequency. A single dominant sequence was then selected to represent each of the accident classes for detailed modeling. In this way, the containment response to the complete spectrum of accidents contributing to the core damage frequency could be evaluated.

Table 8.3-1 of Section 8 of NEDO-33201 Revision 1 identifies the sequences that were used to represent each accident class. The "core damage sequence descriptor" used in the table derives from the results of the Level 1 analysis. Table 7.2-3 of Section 7 of NEDO-33201 Revision 1 identified the sequences, which were significant contributors to the core damage frequency. The representative sequences shown in Table 8.3-1 are based on the Level 1 results presented in Table 7.2-3. For example, Table 7.2-3 indicates that about 99% of the Class I frequency is associated with loss of preferred power (T-LOPP) or loss of feedwater (T-FDW) sequences. From the perspective of modeling the containment response to a severe accident, both of these sequences can be represented as a transient with loss of injection "T-nIN". A similar approach was used in selecting the representative sequence for the other accident classes. Table 8.3-1 provides a summary description for each representative sequence.

Table 8.3-2 couples the representative core damage sequence with one of the release categories illustrated on the containment system event tree, Figure 8.2-1. The resulting scenario is assigned a "containment response sequence descriptor" to summarize the core damage and containment release information. Recalling that Table 8.2-2 provided the total contribution of all accident classes to each release category frequency, Table 8.3-2 provides additional information by presenting the release category frequency in terms of the contribution from each accident class. As indicated in the table, there is a negligible probability of a core damage sequence resulting in overpressure or bypass failure. However, such hypothetical scenarios are retained for evaluation in Section 9 of NEDO-33201 Revision 1 to assure that a conservative source term is developed.

**Representative Severe Accident Sequence**

A representative severe accident sequence in ESBWR is Class I sequence with Reactor Pressure Vessel (RPV) failure at low pressure. Accident Class I involves sequences in which the RPV fails at low pressure; this accident class represents approximately 98% of the core damage frequency. As indicated in Tables 7.2-3 and 7.2-5 of Section 7 of NEDO-33201 Revision 1, the class is dominated by transient sequences in which there is no core injection. Thus, the sequence "T_nIN" is the representative severe accident sequence in ESBWR and it is described below.

**Sequence T_nIN_TSL**

The initiating event for the **T_nIN** sequence is a transient initiated by a loss-of-preferred power. The following events are assumed to occur. Scram is successful. Initial water drop causes RPV water level to go below the trip set point. Both Control Rod Drive (CRD) Pumps fail to restore water before the ADS timer expires, or the injection valves of more than 1 Isolation Condenser System (ICS) train fail to open. The Automatic Depressurization System (ADS) is actuated successfully at about 47 Sec. Depressurization causes ICS to be ineffective. No short or long-term coolant injection to the RPV by the feedwater, CRD or Gravity Driven Cooling System (GDCS) is available. The Automatic Depressurization System (ADS) functions successfully to reduce the RPV pressure. As stated earlier, heat removal by the isolation condensers is not credited.

The pressure in the containment increases as the drywell is filled with steam and heats up. This leads to further heating and pressurization of the drywell air space. Under the "no core injection" scenario, water injection systems fail to provide water to RPV that eventually leads to uncovering of core. About 30 minutes into the event, core uncovery occurs that leads to fuel rod heatup and melting. Fission products and non-condensable gases are swept into the containment through the Depressurization Valves (DPVs) as the core melts. Core heats up from the decay heat and the fuel rods melt. The heat from the melted fuel eventually raises the temperature to more than the melting point of stainless steel core support plate. Melting of core support plate results in loss of core support. This leads to core melt relocation into the lower plenum of RPV.

The core melt in the lower plenum heats up and melts the welds that support the control rod drives and instrumentation tubes at the bottom of RPV. Thus, low pressure RPV failure occurs at the bottom of RPV and core melt debris drops down into the lower drywell floor where Lower Drywell (LDW) water level is zero or low.

The reactor pressure vessel lower head penetrations fail about 6.4 hours into the event. Corium is deposited on the lower drywell floor, which results in local temperatures that are high enough to cause the GDCS deluge line to open. The heat from core debris on the lower drywell floor heats the floor is detected by the embedded thermocouples in the LDW floor. The thermocouples sense presence of core melt on the floor through high temperature and activate the GDCS deluge lines control to open the deluge line valves. As a result, the GDCS pool water drains into the lower drywell and covers the debris bed. The water from the GDCS deluge lines effectively provides water to BiMAC to cool the debris in LDW from both the top and bottom sides. A

successful BiMAC operation leads to a stable cooling state of the debris. Because the debris is quenched by the successful GDCS deluge and BiMAC function, significant core-concrete interaction does not occur. Therefore, no significant fission product aerosols or non-condensable gases are generated in the ex-vessel phase of the accident sequence.

The decay heat from the debris is removed by BiMAC's passive cooling system. The resulting steam moves into the drywell, which is cooled by the PCCS. The condensate water from PCCS returns to GDCS pools and subsequently moves back to BiMAC through the open deluge lines. Thus, Containment heat removal in the short-term is accomplished by successful Passive Containment Cooling System (PCCS) functioning; PCCS pool makeup is successful, thus allowing long-term containment heat removal. The GDCS deluge system and BiMAC are available for debris bed cooling. With successful containment isolation, vapor suppression and containment heat removal the containment remains intact; Technical Specification Leakage (TSL) is the only mode of fission product release.

The key in-vessel events are summarized in Tables RAI-19.2-3-1, -2 and –3. The key ex-vessel events of the sequence are summarized in Table 8.3-3 of NEDO-33201 Revision 1. Figures 8.3-1a through 8.3-1e of Section 8 of NEDO-33201 Revision 1 show the system behavior throughout the accident sequence.

Continued heating by debris of the water in the lower drywell leads to the temperature in the overlying water pool reaching saturation. Steam generation in the lower drywell then leads to further increases in the containment pressure until the PCCS heat removal capacity becomes consistent and comparable to the decay heat generated by the core debris. The containment pressure reaches about 0.65 Mpa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

**Sequence T_nIN_nCHR_FR**

Sequence T_nIN_nCHR_FR is the same as the representative Class I sequence T_nIN, except that the containment response differs because containment heat removal has failed. As a result, containment pressurization increases and controlled venting may be implemented to limit the pressure rise and control the radionuclide release point. Specific guidance for the use of the suppression pool vent has not been developed. Indeed, as discussed earlier, venting in the ESBWR does not appear necessary to limit the containment pressure to less than its ultimate strength in the 24-hour period after core damage. The venting scenario is evaluated here to provide insights into vent timing and provide a basis for the FR release category used in the source term evaluation.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-2a through d show the system behavior throughout the accident sequence. The sequence proceeds as discussed in the previous section except that venting is assumed to occur when the containment pressure

reaches 90% of the ultimate containment strength. As indicated, in Figure 8.3-2b, the drywell pressure has reached less than 70% of the ultimate containment strength within 24 hours after onset of core damage; thus venting would not likely be implemented in this time frame. The 90% assumption is met at 32.3 hours, which is about 2.7 hours before containment over-pressurization would occur if controlled venting were not implemented.

The sequence demonstrates that venting is not required to prevent containment failure in the 24-hour period after onset of core damage due to a Class I event, even if containment heat removal were unavailable. In such a scenario, there is a long time period after core damage to prepare for venting and take other mitigating actions.

**Severe Accident in ESBWR vs ABWR**

GE's approach to severe accidents for both ABWR and ESBWR focuses on satisfying applicable regulatory requirements that provide conservative general design criteria to ensure defense-in-depth against the occurrence of severe accidents and their progression. Both, ESBWR and ABWR demonstrate low Core Damage Frequency (CDF) of 3.8E-08 (Reference – Section 7 of NEDO-33201 Revision 1) and 1.6E-07/year (Reference – GE ABWR FSER 19.1.3.8.3) respectively as compared to NRC goal for CDF of less than 1.0E-05. Both designs provide measures to further assure that the containment function will be preserved to provide investment protection. Although the detailed designs of ESBWR and ABWR are different, the severe accident analyses demonstrate the NRC goal for large release of 1.0E-06/year is met with satisfactory margins. Both analyses show the results for a large release of the order of 1.0E-09/year (Reference – ABWR FSER page 19-34, and ESBWR Section 10 of NEDO-33201 Revision 1). The NRC goal of Conditional Containment Failure Probability (CCFP) to be less than 0.1 was met by ABWR (as shown in ABWR FSER page 40-41). The CCFP calculation for ESBWR is described in Section 8 of NEDO-33201 Revision 1 that demonstrates that the calculated value of CCFP of 0.03 and it meets the NRC goal of less than 0.1.

Table RAI-19.2-3-4 provides a high-level comparison and contrast between the ESBWR severe accident threats and accident mitigation response to that of a comparable accident sequence in the advanced boiling water reactor (ABWR). As the detail designs of ABWR and ESBWR are very different, lower level comparison is not very meaningful.

**DCD/LTR Impact**

No DCD or LTR changes will be made as a result of this RAI response.

| Table RAI-19.2-3-1<br>Summary of Key In-Vessel Events | |
|---|---|
| **Event Description** | **Time (Sec)** |
| Time of Core Uncovery | 1700 |
| Time of Maximum CORE TEMP. >TSAT+500K | 2082 |
| Time of Maximum CORE TEMP. > 2499K | 3967 |
| Time of 1st Relocation to Lower Plenum of RPV | 7262 |
| Time of First Vessel (RPV) Fail | 23069 |

## Table RAI-19.2-3-2a
## RPV and Containment Pressure and Temperature
## And In-Vessel Core Nodal Temperature Profile, in °K at Time = 0.0 Sec

Core Power : Rated Power (Core Covered with Water)
Pressures: RPV(average) = 7.2 MPa , Upper Drywell = 0.11 MPa, Lower Drywell = 0.11 MPa
Wetwell Air Space = 0.11 MPa
Temperatures: RPV(average) = 560 °K, Upper Drywell = 319 °K , Lower Drywell = 319 °K
Wetwell Air Space = 316 °K

| Axial Node | Radial Ring 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|
| 13 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 |
| 12 | 7.63E+02 | 7.54E+02 | 7.29E+02 | 6.85E+02 | 6.40E+02 | 5.61E+02 |
| 11 | 9.46E+02 | 9.32E+02 | 8.93E+02 | 8.18E+02 | 7.27E+02 | 5.61E+02 |
| 10 | 1.00E+03 | 9.83E+02 | 9.51E+02 | 8.91E+02 | 7.64E+02 | 5.61E+02 |
| 9 | 9.76E+02 | 9.58E+02 | 9.37E+02 | 9.02E+02 | 7.67E+02 | 5.61E+02 |
| 8 | 9.79E+02 | 9.63E+02 | 9.49E+02 | 9.22E+02 | 7.81E+02 | 5.61E+02 |
| 7 | 9.49E+02 | 9.40E+02 | 9.33E+02 | 9.16E+02 | 7.76E+02 | 5.61E+02 |
| 6 | 9.12E+02 | 9.10E+02 | 9.10E+02 | 9.04E+02 | 7.63E+02 | 5.61E+02 |
| 5 | 8.72E+02 | 8.70E+02 | 8.77E+02 | 8.86E+02 | 7.45E+02 | 5.61E+02 |
| 4 | 8.17E+02 | 8.09E+02 | 8.25E+02 | 8.54E+02 | 7.18E+02 | 5.61E+02 |
| 3 | 7.29E+02 | 7.22E+02 | 7.37E+02 | 7.54E+02 | 6.63E+02 | 5.61E+02 |
| 2 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 |
| 1 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 | 5.61E+02 |

Notes: (1) Core node temperatures are from top of the core to bottom, inside to outside. (2) The outside ring represents the core/barrel/baffle; the top row represents the non-fueled region at the top of the fuel pin, includes the upper gas plenum, and contains no UO2. (3) Nodes 3 to 12 and Ring 1 to 5 represents active fuel region of the core. (4) Nodal temperature values with " *" denote that the node blocked.

**Table RAI-19.2-3-2b**
**RPV and Containment Pressure and Temperature**
**And In-Vessel Core Nodal Temperature Profile, in °K at Time = 1700 Sec**

Core Power: Decay Heat Power (Core Covered with Water)
Pressures: RPV(average) = 0.25 MPa , Upper Drywell = 0.23 MPa, Lower Drywell = 0.23 MPa
Wetwell Air Space = 0.21 MPa
Temperatures: RPV(average) = 412 °K, Upper Drywell = 408 °K , Lower Drywell = 378 °K
Wetwell Air Space = 328 °K

| Axial Node \ Radial Ring | 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|
| 13 | 4.00E+02 | 4.00E+02 | 4.00E+02 | 4.00E+02 | 4.00E+02 | 4.84E+02 |
| 12 | 4.10E+02 | 4.10E+02 | 4.09E+02 | 4.08E+02 | 4.07E+02 | 4.82E+02 |
| 11 | 4.14E+02 | 4.14E+02 | 4.13E+02 | 4.11E+02 | 4.09E+02 | 4.75E+02 |
| 10 | 4.15E+02 | 4.15E+02 | 4.14E+02 | 4.13E+02 | 4.10E+02 | 4.68E+02 |
| 9 | 4.15E+02 | 4.15E+02 | 4.14E+02 | 4.13E+02 | 4.10E+02 | 4.48E+02 |
| 8 | 4.15E+02 | 4.15E+02 | 4.14E+02 | 4.14E+02 | 4.10E+02 | 4.23E+02 |
| 7 | 4.14E+02 | 4.14E+02 | 4.14E+02 | 4.14E+02 | 4.10E+02 | 4.12E+02 |
| 6 | 4.14E+02 | 4.13E+02 | 4.13E+02 | 4.13E+02 | 4.10E+02 | 4.08E+02 |
| 5 | 4.13E+02 | 4.13E+02 | 4.13E+02 | 4.13E+02 | 4.09E+02 | 4.05E+02 |
| 4 | 4.11E+02 | 4.11E+02 | 4.11E+02 | 4.12E+02 | 4.09E+02 | 4.04E+02 |
| 3 | 4.09E+02 | 4.09E+02 | 4.09E+02 | 4.10E+02 | 4.07E+02 | 4.04E+02 |
| 2 | 4.01E+02 | 4.01E+02 | 4.01E+02 | 4.01E+02 | 4.01E+02 | 4.03E+02 |
| 1 | 4.01E+02 | 4.01E+02 | 4.01E+02 | 4.01E+02 | 4.01E+02 | 4.01E+02 |

Notes: (1) Core node temperatures are from top of the core to bottom, inside to outside. (2) The outside ring represents the core/barrel/baffle; the top row represents the non-fueled region at the top of the fuel pin, includes the upper gas plenum, and contains no UO2. (3) Nodes 3 to 12 and Ring 1 to 5 represents active fuel region of the core. (4) Nodal temperature values with " *" denote that the node blocked.

**Table RAI-19.2-3-2c**
**RPV and Containment Pressure and Temperature**
**And In-Vessel Core Nodal Temperature Profile, in °K at Time = 3600 Sec**

Core Power : Rated Power (Core Covered with Water)
Pressures: RPV(average) = 0.22 MPa , Upper Drywell = 0.22 MPa, Lower Drywell = 0.22 MPa
Wetwell Air Space = 0.22 MPa
Temperatures: RPV(average) = 466 °K, Upper Drywell = 397 °K , Lower Drywell = 375 °K
Wetwell Air Space = 332 °K

| Axial Node \ Radial Ring | 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|
| 13 | 5.87E+02 | 5.81E+02 | 5.67E+02 | 5.42E+02 | 5.15E+02 | 4.90E+02 |
| 12 | 1.44E+03 | 1.40E+03 | 1.30E+03 | 1.15E+03 | 9.87E+02 | 4.92E+02 |
| 11 | 2.11E+03 | 2.04E+03 | 1.88E+03 | 1.60E+03 | 1.27E+03 | 4.94E+02 |
| 10 | 2.228E+03* | 2.15E+03 | 2.10E+03 | 1.86E+03 | 1.41E+03 | 4.90E+02 |
| 9 | 2.14E+03 | 2.13E+03 | 2.04E+03 | 1.91E+03 | 1.43E+03 | 4.77E+02 |
| 8 | 2.14E+03 | 2.14E+03 | 2.10E+03 | 1.99E+03 | 1.50E+03 | 4.65E+02 |
| 7 | 2.14E+03 | 2.07E+03 | 2.04E+03 | 1.96E+03 | 1.51E+03 | 4.58E+02 |
| 6 | 2.03E+03 | 1.99E+03 | 1.97E+03 | 1.94E+03 | 1.53E+03 | 4.52E+02 |
| 5 | 1.87E+03 | 1.87E+03 | 1.90E+03 | 1.90E+03 | 1.46E+03 | 4.41E+02 |
| 4 | 1.72E+03 | 1.71E+03 | 1.75E+03 | 1.86E+03 | 1.25E+03 | 4.31E+02 |
| 3 | 1.58E+03 | 1.55E+03 | 1.59E+03 | 1.61E+03 | 9.68E+02 | 4.18E+02 |
| 2 | 1.03E+03 | 9.80E+02 | 1.10E+03 | 1.14E+03 | 4.40E+02 | 4.04E+02 |
| 1 | 4.14E+02 | 4.11E+02 | 4.14E+02 | 4.14E+02 | 3.99E+02 | 3.97E+02 |

Notes: (1) Core node temperatures are from top of the core to bottom, inside to outside. (2) The outside ring represents the core/barrel/baffle; the top row represents the non-fueled region at the top of the fuel pin, includes the upper gas plenum, and contains no UO2. (3) Nodes 3 to 12 and Ring 1 to 5 represents active fuel region of the core. (4) Nodal temperature values with " *" denote that the node blocked.

**Table RAI-19.2-3-2d**
**RPV and Containment Pressure and Temperature**
**And In-Vessel Core Nodal Temperature Profile, in °K at Time = 7265 Sec**

Core Power: Decay Heat, Core Uncovered with Water,
Core Melt 1st Relocation to Lower Plenum of RPV
Pressures: RPV(average) = 0.27 MPa , Upper Drywell = 0.27 Mpa, Lower Drywell = 0.27 MPa
Wetwell Air Space = 0.26 MPa
Temperatures: RPV(average) = 803 °K, Upper Drywell = 450 °K , Lower Drywell = 382 °K
Wetwell Air Space = 338 °K

| Axial Node \ Radial Ring | 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|
| 13 | 1.00E+02 | 1.00E+02 | 1.00E+02 | 1.00E+02 | 1.00E+02 | 7.66E+02 |
| 12 | 1.000E+02* | 1.00E+02 | 1.00E+02 | 1.00E+02 | 1.00E+02 | 8.48E+02 |
| 11 | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.00E+02 | 1.000E+02* | 9.02E+02 |
| 10 | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 9.26E+02 |
| 9 | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 9.54E+02 |
| 8 | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.02E+03 |
| 7 | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.485E+03* | 1.19E+03 |
| 6 | 1.000E+02* | 1.000E+02* | 1.000E+02* | 1.000E+02* | 2.236E+03* | 1.67E+03 |
| 5 | 3.152E+03* | 3.113E+03* | 3.113E+03* | 3.113E+03* | 2.784E+03* | 1.65E+03 |
| 4 | 3.170E+03* | 3.165E+03* | 3.147E+03* | 3.113E+03* | 2.852E+03* | 1.02E+03 |
| 3 | 3.115E+03* | 3.113E+03* | 3.011E+03* | 2.824E+03* | 2.724E+03* | 7.92E+02 |
| 2 | 2.09E+03 | 2.07E+03 | 2.05E+03 | 1.90E+03 | 1.84E+03 | 5.50E+02 |
| 1 | 1.05E+03 | 1.41E+03 | 1.24E+03 | 1.05E+03 | 1.36E+03 | 4.19E+02 |

Notes: (1) Core node temperatures are from top of the core to bottom, inside to outside. (2) The outside ring represents the core/barrel/baffle; the top row represents the non-fueled region at the top of the fuel pin, includes the upper gas plenum, and contains no UO2. (3) Nodes 3 to 12 and Ring 1 to 5 represents active fuel region of the core. (4) Nodal temperature values with " *" denote that the node blocked.

**Table RAI-19.2-3-2e**
**RPV and Containment Pressure and Temperature at Time = 23069 Sec**

Core Power: Decay Heat, Core Uncovered with Water, Melting Point of UO2 Reached
Vessel Fails at Low Pressure and Core Melt Moves from RPV Lower Plenum to LDW
Pressures: RPV(average) = 0.40 MPa , Upper Drywell = 0.40 MPa, Lower Drywell = 0.40 MPa
Wetwell Air Space = 0.40 MPa
Temperatures: RPV(average) = 859 °K, Upper Drywell = 466 °K , Lower Drywell = 439 °K
Wetwell Air Space = 365 °K

In-Vessel Temperature profile is not meaningful, as the core melt has started migrating from RPV lower plenum into the LDW due to vessel failure.

**Table RAI-19.2-3-3**
**Summary of Analysis Results of Representative Severe Accident Sequence (T_nIN)**
(For comparison to other accident sequences, see Table 8.3-3 of NEDO-33201 Revision 1)

| Sequence Descriptor | RPV Depressurization Initiated (Seconds) | Core Uncovered (hours) | Onset of Core Damage (hours)* | RPV Failure (hours) | Deluge Actuated (hours) | Concrete Ablation 24 hrs. after onset of core damage (meters) | Drywell Pressure 24 hrs. after onset of core damage (MPa) | Containment Vent (hours after onset of core damage) |
|---|---|---|---|---|---|---|---|---|
| T_nIN _TSL | 47 | 0.47 | 1.1 | 6.4 | 6.5 | 0.1 | 0.65 | NA |
| T_nIN_nCHR_FR | 47 | 0.44 | 1.2 | 6.6 | 6.6 | 0.1 | 0.9 | >24 |

**Key:**
nCHR: No containment heat removal
nIN: No injection
FR: Filtered release (controlled vent)
TSL: Technical Specification Leakage
NA: Not Applicable
*Time of maximum core temperature > 2499°K

**Table RAI-19.2-3-4**
**Severe Accident Threats and Protection in ABWR and ESBWR**

| Severe Accident Threats Identified | Consequences Mitigation in ABWR Used for Accident Response | | Consequences Mitigation in ESBWR Used for Accident Response | |
|---|---|---|---|---|
| Hydrogen Combustion | Containment Inerting with Nitrogen | | Containment Inerting with Nitrogen | |
| DCH<br>Containment Over-pressurization and Over-heating<br><br>CCI (gas productions, basemat/pedestal ablation)<br><br>FCI (dynamic loading) | ADS and SRV | Suppression Pool and Vents | ADS and SRV | Suppression Pool and Vents |
| | RHR (Decay Heat Removal) | Vacuum Breakers | IC (Isolation Condenser) | PCCS (Passive Containment Cooling System – Decay Heat Removal), Vacuum Breakers |
| | COPS (Containment over-pressure protection) | AC-Independent Water Addition | MCOPS (Manual Containment over-pressure protection) | GDCS (Gravity Driven Cooling Pools |
| | Corium Shield (Sump Protection) | Lower Drywell Flooder System | BiMAC (Sump Protection) | Lower Drywell Flooder System using GDCS Deluge Lines |
| | Drywell, Floor, Pedestal Design | Drywell Temperature Controlled Flooding | Drywell, Floor, Pedestal Design | Drywell Temperature Controlled Flooding of BiMAC in LDW |