GE Energy

David H. Hinds
Manager, ESBWR

PO Box 780 M/C L60
Wilmington, NC 28402-0780
USA

T 910 675 6363
F 910 362 6363
david.hinds@ge.com

MFN 06-313

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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 40 Related to ESBWR Design Certification Application – ESBWR Probabilistic Risk Assessment – RAI Numbers 19.1-8 (b) (Revised Response), 19.1-9, 19.1-10, 19.1-16, 19.1-18, 19.2-4. 19.2-5, 19.2-15, 19.2-16, 19.2-19, 19.2-20, 19.2-21, 19.2-23, 19.2-34, 19.2-58 through 19.2-62, and 19.2-64**

Enclosures 1 and 2 contain GE's response to the subject NRC RAIs transmitted via the Reference 1 letter. Please note that a revised response to RAI 19.1-8(b) previously provided in GE's Reference 2 letter is included.

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

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

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References:

1. MFN 06-222, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 40 Related to ESBWR Design Certification Application,* July 5, 2006
2. MFN 06-257, Letter from David Hinds to U.S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 40 Related to ESBWR Design Certification Application – ESBWR Probabilistic Risk Assessment – RAI Numbers 19.1-8, 19.1-17, 19.2-6, 19.2-10, 19.2-13 and 19.2-18,* August 18, 2006

Enclosures:

1. MFN 06-313 – Response to Portion of NRC Request for Additional Information Letter No. 40 Related to ESBWR Design Certification Application – ESBWR Probabilistic Risk Assessment – RAI Numbers 19.1-8 (b) (Revised Response), 19.1-9, 19.1-10, 19.1-16, 19.1-18, 19.2-4. 19.2-5, 19.2-15, 19.2-16, 19.2-19, 19.2-20, 19.2-21, 19.2-23, 19.2-34, 19.2-58 through 19.2-62, and 19.2-64
2. MFN 06-313 – ROAAM-Review of "Severe Accident Management in Support of the ESBWR Design Certification"

cc: AE Cubbage USNRC (with enclosures)
GB Stramback GE/San Jose (with enclosures)
eDRF 0000-0045-4145

ENCLOSURE 1

MFN 06-313

Response to Portion of NRC Request for
Additional Information Letter No. 40
Related to ESBWR Design Certification Application
ESBWR Probabilistic Risk Assessment
RAI Numbers 19.1-8 (b) (Revised Response), 19.1-9, 19.1-10,
19.1-16, 19.1-18, 19.2-4. 19.2-5, 19.2-15, 19.2-16, 19.2-19, 19.2-20,
19.2-21, 19.2-23, 19.2-34, 19.2-58 through 19.2-62, and 19.2-64

NRC RAI 19.1-8(b)

Full RAI provided for completeness.

Discuss how the operating efficiency of the Passive Containment Cooling System (PCCS) (including thermo-physical properties, heat transfer coefficients, steam condensation efficiency, fission product removal, and axial and radial velocity distribution within the condenser tubes) is impacted by each of the following: (a) large quantities of non-condensible gases such as CO2 and H2, (b) corium-concrete interaction (CCI) – generated aerosols including plugging effects, and (c) increases in Isolation Condenser (IC) pool temperatures as the event progresses. Support the responses with an appropriate analysis for each case.

GE Response

This revised response addresses the part (b) of this RAI.

(b) During severe accident conditions during which PCCS is required to operate, no CCI is possible because of BiMAC; this is why the BiMAC was put in the design. Even under hypothetically assumed CCI scenarios, the lower drywell is flooded with a water height of 10 meters or higher. With such water height, the concentration of CCI generated aerosol above the surface of the water is significantly small (reference NUREG/CR-5901) and does not impact PCCS performance.

NRC RAI 19.1-9

It is traditionally assumed that Reactor Pressure Vessel (RPV) failure occurs at the bottom of the lower head. (All the analyses presented in the Safety Analysis Report (SAR) make this assumption.) However, calculations performed by Oak Ridge National Laboratory (ORNL) for operating BWR designs suggest that early relocation of stainless steel control blades and cladding material could result in alternative failure modes. Discuss the implications of the RPV failing at other locations than the very bottom. Include in this discussion an assessment of the following: (a) the impact of early relation of non-heat-bearing debris to the lower plenum on failure location, and (b) the impact of a change in failure location (i.e., at the level of the lower grid plant) on sequence progression and containment loads. This should include consideration of the impact on fission product release, CCI, steam explosion, and actuation of the cavity flood system.

GE Response

GE made no such assumption, in fact the whole severe accident management approach, and the BiMAC in particular, were developed with full cognizance of such uncertainties.

This impact is not known with any significant degree of confidence, and this is why the mode and location of failure is treated as a splinter set of scenarios (see addenda in full ROAAM review Severe Accident Treatment (SAT) report provided as the "Attachment to the GE's response to RAI 19.2-5"), just as was done in the Mark I liner attack issue resolution work (Theofanous et al, 1991, 1993).

A "high/side" failure such as the one postulated would make all events more benign than they were made out to be in our analyses for the purpose of bounding. The reason is that quantities and rates of melt relocation from the RPV into the LDW would be significantly lower. In particular this would eliminate the DCH and steam explosion threats, and would make all BiMAC-related performance issues even more reliable. On fission product release the impact would be negligible one way or another.

References

T. G. Theofanous, W. H. Amarasooriya, H. Yan and U. Ratnam, "The Probability of Liner Failure in a Mark-I Containment," NUREG/CR-5423, August 1991.

T. G. Theofanous, H. Yan (UCSB); M.Z. Podowski, C. S. Cho (RPI); D. A. Powers, T. J. Heames (SNL); J. J. Sienicki, C. C. Chu, B. W. Spencer (ANL); J. C. Castro, Y. R. Rashid, R. A. Dameron, J. S. Maxwell (ANATECH), "The Probability of Mark-I Containment Failure by Melt-Attack of the Liner," NUREG/CR-6025, November 1993.

NRC RAI 19.1-10

Provide an assessment of the risk (frequency and consequences) associated with a rupture of the pipe carrying non-condensible gases from the PCCS to the suppression pool. (It would appear that this would not only disable the operation of the PCCS, by eliminating the pressure differential, but would also cause the suppression pool to be bypassed and the containment pressure to increase in an unabated manner.) Based on this assessment, either address this failure in the Containment System Event Tree (CSET) or justify its omission.

GE Response

These pipes are not subject to any significant loading at any time during such accidents, so their failure is physically unreasonable, and as a consequence such events need not be part of explicit consideration in the PRA.

NRC RAI 19.1-16

The quantification of loads in the steam explosion calculations is based on a given set of initial and boundary conditions (e.g., melt pouring rate of 720 kg/s, premixing area of ~0.03 m^2, melt volume fraction of 22%, etc.) and certain assumed values of explosion parameters β and γ (see PRA, pages 21.4-7 and 8). Given that there are uncertainties in severe accident progression that provide initial and boundary conditions for explosion calculations, sensitivity analyses would be useful in providing insights into the uncertainties in the loads. The PRA provides only limited sensitivity calculations involving water pool depth. Provide additional sensitivity calculations involving pouring rate, premixing area, subcooling, and the choice of β to confirm that the loads are indeed bounding. Such sensitivity calculations proved useful for addressing uncertainties in steam explosion loads for AP1000 and more recently, as part of the international SERENA (Steam Explosion Resolution for Nuclear Applications) exercise.

GE Response

Precisely so, our choices were informed by all this previous experience in which the authors of the report and the tools utilized were principal participants. The key quantities are melt pour rate and water subcooling, and these were chosen in a bounding fashion as described in the Section 21.4 of NEDO-33201 Rev 1. The steam explosion calculations in the Section 21.4 were carried out with the PM-ALPHA.L-3D, and ESPROSE.m codes. The parameter values are explained in the code verification manuals as cited in Section 21 as Reference 21.4.6-26 for ESPROSE.m (a computer code for addressing the escalation/propagation of steam explosions) and Reference 21.4.6-27 for PM-ALPHA (a computer code for addressing premixing of steam explosions). "Based on the above references, no additional sensitivity studies were performed because any sensitivity studies of these parameters would be utilizing lower values of melt pour rate and less subcooling (warmer water), which would produce even lower loads." Moreover as we show, and explain in the Section 21 of NEDO-33201 report, the margins to pedestal failure from postulated explosions in shallow water pools are great and not a fruitful subject of sensitivity analyses.

NRC RAI 19.1-18

Provide documentation of the analyses of uncertainties and sensitivities for the MAAP ESBWR model application. Discuss the applicability of the extended sensitivity analysis suggested by the Electric Power Research Institute MAAP Users Group (EPRI/MUG) for BWR applications.

GE Response

This RAI was asked in the context of the Level 2 PRA. A review of the 1992 MAAP BWR Application Guidelines (EPRI TR-100742) shows the following statement on Page 1-1 of this report: "These objectives pertain to the thermal-hydraulic behavior within the reactor vessel prior to any core damage." This indicates that the guidelines pertain more to Level 1 PRA. However, the Application Guidelines were reviewed for applicability to the Level 2 PRA.

Sections 1-5 of the Application Guidelines include: Introduction, MAAP Code Overview, MAAP Thermal-Hydraulic Qualification Overview, General BWR Design and Operation, and Discussion of MAAP Modeling.

Section 6, BWR MAAP Application Guidelines, discusses the following sequences:

1) Loss of Injection
2) ATWS
3) Small and Medium LOCA
4) Loss of Containment Heat Removal

The analysis for each of these (Level 1) sequences includes recommended sensitivity analyses. These recommended sensitivity analyses could be characterized as important inputs that should be reviewed and verified or conservative inputs used or inputs studied with sensitivity runs. The important inputs for the 4 sequences discussed include the following:

1. Initial RPV Water Inventory
2. FW operation during the first few seconds of the transient
3. Assumed battery life
4. S/RV effective flow area
5. ECCS flow curves, especially low pressure systems
6. TDSLC - time required to shutdown the reactor after initiation of SLC
7. Break area and elevation
8. Mode of long term vessel makeup during loss of CHR.

The Level 2 Analysis reviews the sequences which dominate core damage. Review of the MAAP Parameter file and the Level 2 analysis shows the following concerning the above inputs:

1. The Initial RPV Water Inventory - verified against the output of WEVOL, an internal GE calculation of RPV weights and volumes.
2. FW operation during the first few seconds - verified against the input used for the DCD Chapter 15 SBO Analysis.
3. Assumed battery life - battery life for the ESBWR batteries (24 hours) is stated in DCD Chapter 8 Section 3.
4. S/RV effective flow area - the effective area in the parameter file was verified using MAAP guidelines for parameter development.
5. ECCS flow curves - for the dominant sequences in the Level 2 analysis, except for the ATWS sequences, no RPV injection is included in the sequences. The ATWS sequences include FW injection, one with FW runback and the other without FW runback, which increases the decay heat produced.
6. TDSLC - the two ATWS sequences evaluated as Class IV sequences in the Level 2 analysis do not credit SLC initiation. There is one ATWS sequence evaluated as a Class I sequence, RPV Failure at Low Pressure. The MAAP analysis for this sequence used a conservative value of 1800 seconds for TDSLC compared to the time to achieve Hot Shutdown after start of SLC Injection, 195 seconds, calculated by TRACG.
7. Break area and elevation - the LOCA sequence included in the dominate CDF sequences is a Medium LOCA sequence. The break is assumed in the GDCS Equalizing line, which is the lowest line break elevation above the core. The supplemental sequences include a Large LOCA which assumes a non-mechanistic break location.
8. Mode of long term vessel makeup during loss of CHR - Loss of CHR Class was evaluated as a supplemental sequence in the Level 2 analysis. Due to the long time available for mitigating actions prior to core damage, inclusion of this Class was not required in the Level 2 Analysis.

The validation of the MAAP model for ESBWR response in design basis accidents is described in a separate report, "MAAP Analysis of ESBWR and Comparison to TRACG", EPRI-TR1011712. This report has previously been provided to the NRC. In addition, the use of MAAP for Level 1 success criteria will be further assessed in the response to RAI 19.1.0-1 concerning thermal-hydraulic uncertainty.

Therefore, the Level 2 analysis is consistent with the MAAP BWR Application Guidelines.

NRC RAI 19.2-4

Discuss how in-vessel recovery of a damaged core would be approached in the ESBWR design, and the use of AC-independent fire water system for this purpose. Justify that the treatment (or lack of treatment) of in-vessel recovery in the Level 2 PRA is appropriate.

GE Response

With a core damage frequency of 10^{-8} per year without counting such actions, and considering the high uncertainties associated with arresting melt in a partially degraded core, there can be no substantial benefit for more analysis on procedures that in any case will be in place.

The description in Section 21 of NEDO-33201 summarizes the In-Vessel Retention (IVR) challenges due to RPV lower vessel penetrations of instrumentation tubes and CRDs.

NRC RAI 19.2-5

A fundamental component of the validity of the Risk-Oriented Accident Analysis Methodology (ROAAM) approach is the quality of the independent peer review. Provide additional detailed information to substantiate that review was independent and comprehensive. This would include the affiliations, qualifications and relevant experience of the reviewers to the area reviewed, an estimate as to level of effort each devoted to the review, the individual directions given regarding the scope and depth of their review, and information as to joint meetings and interviews.

GE Response

While recognizing the limitations, under the time schedule for delivering the initial PRA to the NRC, we had no choice but to conduct a limited ROAAM review that included two independent experts, namely Dr. Robert Henry, of Fauske and Associates, and Dr. Fred Moody, a retired GE employee and now an independent consultant. Now that the time permits we have undertaken a full ROAAM review involving 9 independent experts as shown in Table 1 below:

Table 1 – Summary of Independent Full ROAAM Reviewers

Name	Company	Expert's Review Areas	Efforts
Hans Fauske	Fauske & Associates, Inc. USA	All aspects of BiMAC engineering, steam explosions, and all safety	1 week
Joe Rashid	ANATECH Corp. USA	All structures under thermal and explosive loading	1 week
Peter Griffith	Massachusetts Institute of Technology, USA	Thermal Hydraulic of BiMAC	4 weeks
Fred Moody	Consultant USA	All aspects of BiMAC engineering, DCH, and all safety	4 weeks
Robert Henry	Fauske & Associates, Inc. USA	All aspects of BiMAC engineering, DCH, and all safety	1 week
Brian Turland	Sercoassurance, UK	Steam explosions, and all aspects of Sever Accident Management (SAM)	1 week
Harri Tuomisto	FORTUM Engineering, Finland	All aspects of SAM	1 week
Manfred Fisher	Framatome ANP, Germany	Material interactions and ex-vessel melt behavior	1 week
Masaki Saito	Tokyo Institute of Technology, Japan	High temperature material interactions, melt attack of BiMAC	1 week

All are well known to the USNRC and most have participated in previous ROAAM reviews—brief CV's will be included with the revised report that will document the review. The review is currently under way and it is expected to be completed by the end of September 2006.

The full ROAAM review of Severe Accident Treatment (SAT) report, provided to the ROAAM reviewers, is attached to this RAI letter as the "Attachment to GE Response to RAI 19.2-5". The content of the full ROAAM Severe Accident Treatment (SAT) report is the same as provided in Section 21 of NEDO-33201 Rev 1 except the format of the report is slightly different. The new SAT report addenda pages describe the additional information provided to the full ROAAM reviews and are marked as "unverified page for review" on each page. We expect revised Section 21 of NEDO-33201 to be submitted at the end of October 2006, which will include and incorporate all the full ROAAM reviewers' comments and resolutions.

NRC RAI 19.2-15

If the containment sprays are turned on while the PCCS is removing heat, the resulting drop in drywell pressure may interrupt the flow to the GDCS, and eventually the RPV. Although the scenario progression from this point is not clear, core damage appears possible. In view of the potential risk significance, please provide an assessment of the affect of spray system operation on core cooling. Include in your response (a) a supporting thermal-hydraulic analysis for this event, and (b) a description of system design features, operating procedures, or administrative controls that reduce the likelihood of this operator action.

GE Response

This is a temporary event, as vacuum breaker action will equalize pressures. If the GDCS flow to RPV is temporarily stopped long enough to cause RPV water level to drop to a sufficiently low level, the equalization lines will open to provide water to RPV from the suppression pool to keep the water level in the core above the top of the active fuel and provide core cooling. No analysis is warranted.

NRC RAI 19.2-16

Discuss the rationale for designing the containment over-pressure protection system (MCOPS) as a manually actuated system in ESBWR versus a passively actuated system in ABWR. Address the apparent inconsistency of this approach with the passive design philosophy of ESBWR.

GE Response

In the ESBWR the margins to containment failure are very large, and the time scales for action are very long, so a manual actuation system is deemed to be sufficiently reliable. In fact manual actuation is desirable because the time for venting can be based on plant, weather, and evacuation information available to the operators. Finally, venting is not even credited in the ESBWR PRA for preventing containment failure. It is in the analysis simply to mitigate the magnitude of hypothetical releases resulting from passive containment heat removal failure. Active systems backing up passive functions are consistent with the ESBWR design philosophy.

NRC RAI 19.2-19

The deluge downcomers are presumably headered together at some point to feed into the basemat internal melt arrest and coolability system (BiMAC). Describe how the downcomers/headers are protected from being disabled by "corium splatter", corium jets from an off-center head failure, or a missile consequent to RPV head failure. Describe how such a disabling event would affect subsequent accident progression.

GE Response

In a postulated "high/side" failure locations such as that discussed in RAI 19.1-9, both the quantity of melt and the driving force (low pressure scenario) would be of small magnitude, and thus insufficient to produce damage; even if a postulated "high/side" failure occurred, it would be even more unlikely that it would azimuthally coincide with the position of one of the downcomers. All other failures, and by far the most likely ones, would involve one or more of the lower head penetrations, and thusly they would be directed straight down, with a margin of more than 2 meters from the closest pedestal wall (and BiMAC downcomer attached to it). Such a distance is deemed adequate to exclude any direct interaction between relocating melt and BiMAC downcomers.

NRC RAI 19.2-20

Provide the following information regarding the BiMAC geometry in the lower drywell region: (a) additional information on the shape/configuration of the 20 cm refractory layer (it appears as being 'cone-shaped'), and (b) additional information related to the water-cooling distribution system of BiMAC (e.g., the number of cooling pipes connected to the main header, the spacing/separation between the cooling pipes, etc.).

GE Response

Please see Section 21 of NEDO-33201 Rev 1, where such information is already included. Figures 21.3.2-5, 21.4.2-1, 21.4.2-2, 21.5.2-1a, and Tables 21.5.4.3-1, 21.5.4.3-2, 21.5.4.3-3, 21.5.4.3-4 and 21.5.4.3-5 provide BiMAC geometry, shape, configuration, and water-cooling distribution system.

NRC RAI 19.2-21

Provide additional information regarding coolant flow into the BiMAC device during the initial phase of BiMAC operation (beginning with debris relocation and deluge system actuation, and ending with establishment of natural circulation). This should include flow rates into the distributor versus time from (a) the GDCS, (b) the RPV, and (c) the BiMAC downcomers. Address the potential for local steam starvation and dryout due to countercurrent flow of water from the water pool into the BiMAC channel outlets. Provide the final bounding state (i.e., quasi-steady state) of the core debris within the lower drywell.

GE Response

The concern expressed here is about counter-current flow and its effects on BiMAC performance. What is missed in this concern is that the system is self-adjusting, in that flow intensity and direction will depend on the amount and distribution of heat on the BiMAC channels. In a completely uniform final state all channels will run forward, and water supply will be from the downcomers (the GDCS initially until the pools are emptied). This constitutes the final bounding state used to size the BiMAC channels, downcomers, and deluge lines (see Section 21.5 on Basmat Melt Penetration of NEDO-33201 Rev 1). In reality we would expect that some or none of the BiMAC will be thermally loaded because, as explained in Section 21 of NEDO-33201 Rev 1, much of the core debris will be released late and slowly, into a deep water pool, thus undergoing quenching, and consequently any decay power in it not being part of the heat balance on any molten portion that would thermally load the BiMAC. In such a case those BiMAC channels that are not, or only slightly, loaded will run in reverse. Note that even in the final bounding state (based on the whole amount of the core) the thermal loads are very modest, and it would take pipes that are one-fourth in cross-sectional area of that selected to create conditions that might lead to momentary flow starvation. The results of these calculations are included in the full ROAAM review Severe Accident Treatment (SAT) report and its addenda provided as an attachment to the GE response to RAI 19.2-5. All these trends will be demonstrated with the BiMAC-systems tests at ½-scale, as well as by the single-channel tests at full scale.

NRC RAI 19.2-23

As the PRA correctly points out (page 21.5-3), the effectiveness of the BiMAC design needs to be confirmed through testing. Until such data is available, the performance of BiMAC cannot be relied upon with a high degree of confidence. Describe the tests and analyses that will be performed to support design certification and/or issuance of a COL, who will perform these tests, and when the results of the tests will be submitted. Describe the test program planned to ensure reliable and predictable operation of the BiMAC device, including whether these tests will involve single and/or multiple cooling channels, and the anticipated scale of the tests.

GE Response

The BiMAC test program was initiated July 2006. The program is oriented not only in demonstrating that the choices made (diameter, angle of inclination) will yield a robust performance, but also in directly determining the fragilities to local and global thermal loading. To this end the program will include component tests at full scale and systems tests at half-scale. The component tests will be conducted with single pipes, at full scale (5 m long, 10 cm in diameter) and with prototypical materials (steel pipes, water coolant). In these tests we will vary the angle of inclination, and the magnitude and location of a high thermal load needed to cause burnout, while on the rest of the pipe we will apply nominal thermal loads that are related to the final bounding state. In addition, looking for the fragility to local heating we will run tests with 5 cm in diameter, 5 m long pipes. Our analysis indicates that at this diameter we can expect some degradation of performance. Depending on the results we may pursue a third diameter, still at full length, until we find clear indication of degradation in performance. The system's tests will be run at one-half geometric scale, that is 5 cm in diameter, 2.5 m long pipes, and they will involve a near ¼-symmetry portion of the whole BiMAC, that is about 20 to 30 pipes. In these tests we will look for flow interaction effects between channels under uniform and non-uniform heating, and will apply local heating that is to represent peaking of heat fluxes as described in Section 21 of NEDO-33201 Rev 1. A conceptual design report will be issued by 10-01-06, the final design report including shakedown test results will be issued by 02-01-07, and the final BiMAC Thermal-Hydraulic Tests report is scheduled for 08-01-07. There will be opportunities for the USNRC to visit the facility and comment on the test plan.

NRC RAI 19.2-34

The model used to quantify DCH loads in Section 21.3.4.3 does not appear to have an explicit term for the contribution of combustion of hydrogen with residual oxygen in the drywell atmosphere. Provide a discussion of the basis for this omission.

GE Response

It is correct that this effect was not included. Because of the quantities and gas compositions involved, combustion is not the proper characterization. In fact due to the prolonged release of hydrogen into the containment during the core melt and relocation, any tendency to react with residual oxygen would have already taken place well before a postulated DCH event. To the extent that such residual reactions would remove non-condensable gases, they may even be beneficial in reducing the loading during the DCH event.

NRC RAI 19.2-58

In PRA, Section 21.3.4.4, GE described an analysis to address the liner integrity for temperatures greater than 1000°K, using LS-DYNA3D. In the GE model, a piece of liner between a neighboring set of anchors and the presence of concrete behind the liner was considered. GE showed in Figures 21.3. -22 that the resulting maximum effective plastic strains in the liner between anchors at temperatures 1400°K and 1650°K are 1.4% and 7.26%, respectively. In Section 21.3.4.3, GE stated that the drywell pressure is predicted to be around 6 bars (0.6Mpa). However, it is not obvious that the pressure load is included in the LS-DYNA 3D model. Provide:

a) the material models for both liner and concrete at high temperature used in LS-DYNA3D model, including stress-strain relation and strain rate effect.

b) a discussion of the effect of high temperature degradation on the ability of the liner and concrete to resist the pressure load.

GE Response

First it should be noted that as stated in Section 21 of NEDO-33201 Rev 1, the code used is not the commercial version LS-DYNA3D but the LLNL code DYNA3D. This is a much advanced code, with unique capabilities in the areas of interest here, including explosive loads and/or high temperatures.

a) Material properties for the liner were provided in Figure 21.3.4.4.2 of Section 21 of NEDO-33201 Rev 1. Concrete properties are not relevant, as the time scale is way off the time scale needed to affect concrete wall structural stability.

b) The high internal pressures have no bearing on the concrete, which remains structurally intact. The liner, backed by the concrete, will deform under thermal stresses, and Section 21 of NEDO-33201 Rev 1 shows that this deformation is accommodated by creep, while remaining at strain levels well below what might be considered as potentially threatening.

NRC RAI 19.2-62

In PRA, Section 21.4.4.4, GE described the structural response analyses for the pedestal and the BiMAC device subjected to EVE pressure impulses. The K&C model (Karagozian and Case) was used for concrete and rebars included in the model. The pressure impulse loads analyzed range from 200 kPa-s to 600 kPa-s. The impulse loads are characterized as high frequency loads and, therefore, strain rate effect on material properties is expected to be important. Provide:

a) a description of how the strain rate effect is considered for both concrete and steel material models (material properties are typically obtained from pseudo static tests (low cyclic));

b) a detailed description of the K&C model;

c) a description of how the reinforced concrete pedestal is modeled in the LS-DYNA3D model;

d) a description of how the failure of the pedestal impacts the RPV supports, which are structurally supported by the pedestal.

GE Response

a) and b) The model used is not the commercial LS-DYNA3D but LLNL's own code developed and verified for explosive loads, as described in reference "Noble, C. J. et al (2005). "Concrete Model Description and Summary of Benchmark Studies for Blast Effects and Simulation", UCRL-215024, Lawrence Livermore National Laboratory (July, 2005)" given in Section 21.4 of NEDO-33201 Rev 1. This reference contains all descriptions requested under a) and b).

c) This is already given in Section 21.5 of NEDO-33201 Rev 1 (Figures 21.5.4.4-2 and related text).

d) As noted already in Section 21.5 of NEDO-33201 Rev 1, SE loads on the pedestal are localized and any hypothesized failure would be local, having no impact on the pedestal as a whole, and reactor-bearing capacity.

NRC RAI 19.2-64

In PRA, Revision 1, Section 21.4.5, GE made a statement that the reactor pedestal and BiMAC structural designs are capable of resisting explosion load impulses of magnitudes in the 100's of kPa-s. Provide the technical justification, including failure criteria used, for this statement.

GE Response

This question refers to the now obsolete Rev 0. The information requested is provided in Section 21 of NEDO-33201 Rev 1 (separate submittal of Section 21 in December, 2005). The Section 21.4.4.4 describes the quantification of fragility for pedestal as well as BiMAC pipes based on DYNA3D model (e.g., see Figures 21.4.4.4-2 through 21.4.4.4-6 for DYNA3D analyses results).