From: Getachew Tesfaye
Sent: Wednesday, May 14, 2008 4:32 PM
To: 'Pederson Ronda M (AREVA NP INC)'
Cc: John Rycyna; Joseph Colaccino; Hossein Hamzehee; Lynn Mrowca; Edward Fuller; Hanh Phan; Theresa Clark; Jim Xu
Subject: Draft RAI No. 6
Attachments: Draft RAI 6 SPLB 266 & 277.doc

Ronda,
Attached please find the subject draft RAI. If you have any question or need clarifications regarding this RAI, please let me know as soon as possible, I will arrange a telecon with our technical staff. Our staff has requested that the telecon be held no later than **Wednesday May 21, 2008**.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP

E-mail Properties

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Subject: Draft RAI No. 6
Sent Date: 5/14/2008 4:11:44 PM
Received Date: 5/14/2008 4:31:00 PM
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Files	Size	Date & Time
MESSAGE	88561	5/14/2008
Draft RAI 6 SPLB 266 & 277.doc	75914	

Options
Expiration Date:
Priority: olImportanceNormal
ReplyRequested: False
Return Notification: False

Sensitivity: olNormal
Recipients received:

DRAFT
Request for Additional Information No. 6, Revision 0
5/14/2008
U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19.1
SPLB Branch

QUESTIONS

19-78

Severe accident "relevant" accident scenarios are defined as those having a PRA Level 1 CDF greater than a single specified frequency. Please provide information on the rationale for selecting this scenario selection criterion, including a discussion of how these scenarios would bound various scenarios that could have significantly different levels of consequences and public risk factors.

19-79

Please provide the description, technical basis, and decomposition of phenomena pertaining to induced rupture of the reactor system pressure boundary during severe accidents.

- Provide the analyses for hot leg and surge line rupture. Include base-case analyses and sensitivity studies. Discuss the consequences of hot leg failure, the impact of material properties, the sensitivity of phenomena to natural circulation flow rates, the impacts of different initiators, and the effects of small breaks or seal leaks. Also identify the representative transients.

- Provide the analyses for steam generator tube rupture. Include base case analyses and sensitivity studies. Discuss the impact of degraded tubes, from either wear at anti-vibration bars, wear from foreign objects above the tube support plate, or from stress corrosion cracking (note that the tubes will be fabricated from Inconel 690). Include considerations of pressure-induced rupture upon secondary side depressurization prior to core damage, and high temperature-induced creep rupture after core damage.

- Provide a summary of the relevant MAAP cases and key results, including failure times and/or damage fractions, plots of hot leg and steam generator tube temperatures, plots of natural circulation flow rates (including core-to-upper plenum, countercurrent and unidirectional flow through the hot legs, and countercurrent and unidirectional flow through the steam generator tubes).

- What are the total probabilities of induced primary system rupture for the spectrum of break sizes considered?

- Describe how the core damage end states were defined. Show the methodology used for applying these to the Level 1 core damage sequences. For the core damage end state TP, please explain the basis for calculating the times to hot leg or surge line rupture, steam generator tube rupture, and vessel failure.

19-80

Please discuss any modifications to the MAAP 4 code that are associated with the core catcher, the heavy reflector, or the severe accident depressurization valves, that were not discussed in the topical report ANP-10268P.

19-81

For the Level 2 phenomenological basic events (i.e., those events with identifiers "L2PH…"), please supply the associated discrete uncertainty probability distributions and the supporting bases for these values. Also supply the CET-supporting fault trees that utilize these events. Examples of such basic events would be MCCI probabilities in various circumstances, containment overpressure failure probability during debris quench, in-vessel and ex-vessel steam explosion consequences, and "rocket" mode RPV failure. In addition, please provide the numerical values for any other branch probabilities of events in the containment event trees documented in Appendix 19C of the FSAR that are not covered by the above request.

19-82

Please define the MAAP cases run to support the CET quantification and phenomenological evaluations in the Level 2 PRA. For each initiator, indicate the status of safety injection, feedwater, secondary depressurization, primary depressurization, SAHRS, PARs/combustion, and other requirements. Explain the principal reason for including the case and identify the containment failure mode, release category, and sensitivity case. Include other relevant scenario details.

19-83

Please provide a source term grouping diagram that includes the various attributes of the accident sequences that have been considered in defining and describing the release categories. In addition, please provide a mapping of sequences simulated by MAAP 4.0.7 runs to the release categories.

19-84

The tabulated release fractions in the FSAR show peculiarities as related to reactor coolant system retention, and overall release magnitudes. In order to confirm the consistency of the tabulated data, please provide the following source term information for the Level2/Level 3 interface:

- The technical bases for various release fractions listed in Table 19.1-20 (i.e., release evolution of various groups from fuel, retention in the reactor coolant system, the steam generator secondary side [in case of SGTR], the reactor containment, and the annulus and/or other buildings, as applicable). Please limit the information to risk- and consequence-dominant scenarios.
- Show how the twelve MAAP radiological groups (defined in pages 19.1-91 and 19.1-92) were regrouped into the nine radiological groups in Table 19.1.-20 of the FSAR.
- For each of the Release Categories listed in Table 19.1-20, provide the release characteristics (e.g., time of alarm, delay time, number of plumes, plume duration, plume energy, etc.) necessary for ex-plant consequence analysis.
- It is recognized that an assessment of uncertainties in source terms was not performed. Instead, a number of MAAP parametric sensitivity calculations were performed. Please provide a list of sensitivity cases, including the MAAP parameters, the associated ranges, the basis for their selection, and the resulting impact on the calculated fission product release and transport results as applicable to U.S. EPR.

- Provide the assumptions related to the number of steam generator tubes that are considered in the MAAP analyses that result in the release quantities listed for Release Category 702 of Table 19.1-20 (i.e., steam generator tube rupture with no scrubbing). In addition, provide the technical justification that supports the bounding nature of the selected scenario for this release category with regards to the multiple tube ruptures in the steam generators under this accident conditions.
- Please provide the technical bases for apparent high retention associated with the release magnitude for Release Category 802 (i.e., Interfacing Systems LOCA with no scrubbing). In addition, please discuss the rationale that the selected scenario for this release category envelops other Interfacing Systems LOCA scenarios.
- The data listed in various tables are not self-explanatory, since not all abbreviations and/or acronyms have been fully defined. For instance, Table 19.1-26 lists fifteen different acronyms for the core damage end states (CDES), which are not described. Please provide a complete list of abbreviations and acronyms applicable to FSAR Sections 19.1 and 19.2. In addition, with respect to this table, please discuss the differences between the two TP1 entries, and provide the contributions of each CDES to the applicable release category.

19-85

Please provide the analysis of the scrubbing of releases in the steam generator for release categories 701 and 702.

19-86

Please provide the analysis of the Interfacing System LOCA release category 802 results reported in Table 19.1-20 of the FSAR. Include the MAAP analysis results, and show the details of the fission product deposition in the fuel building and in the safeguards building. Include the analysis showing flooding levels for the break scenarios considered.

19-87

Please provide the assessment of the potential for in-vessel retention of core debris for the various core damage end states. Include a discussion of the potential for recovery during the following phases: core heatup to the onset of core melt; the onset of core melt to relocation of core debris into the lower head of the vessel; and relocation into the lower head of the vessel until vessel failure. Please relate the probability of successful quenching in each phase to the time when the primary system is depressurized. Provide the rates of hydrogen production from debris quenching during each phase. Please include results from applicable MAAP runs.

19-88

One of the uncertainties that can potentially impact the quantification of a number of severe accident phenomenological issues in the containment event progression is the expected mode of in-vessel crucible crust failure and melt relocation to the lower plenum. Please provide the technical bases for the quantification of this uncertainty in the U.S. EPR PRA. Furthermore, please discuss the details of the quantification process of accounting for the presence of the heavy reflector near the core boundary (a significant point of difference with conventional U.S. PWR designs).

19-89

In the probabilistic analysis of vessel rocketing at the time of vessel melt-through, please provide the technical basis for quantification (or exclusion from the analysis, if not quantified) of the following forces:

- Jet forces of corium and RCS gases;
- Pressure differential between the reactor pit and the upper containment, including the effect on reactor pit pressure of blowdown, hydrogen combustion, and/or direct heat transfer from ejecting debris and pit atmosphere; and
- Vessel restraining forces, including the effect of uncertainties on pipe temperature at the RPV nozzles and its impact on yield strength, if applicable.

19-90

The U. S. EPR Final Safety Analysis Report, Pages 19.1-67 through 19.1-70 discuss the probabilistic evaluation of the potential impact of in-vessel and ex-vessel steam explosions. The estimated probability of containment failure due to an in-vessel steam explosion for high-pressure scenarios is by a factor of ~4 greater than the corresponding probability for low-pressure scenarios. This is appears to be counter- intuitive because the likelihood of triggering a steam explosion increases as the pressure is reduced. Please provide and discuss:

- The approach and the quantification process that support the estimated probabilities of containment and lower head failure.
- The range and uncertainties associated with the pour mass, composition, temperature, and location (relative to the reactor cavity wall), for assessment of the probability of containment failure due to ex-vessel steam explosions.
- The range and uncertainties associated with the water pool depth and temperature for the assessment of the probability of containment failure due to ex-vessel steam explosions.
- The potential impact of ex-vessel steam explosions on containment integrity (e.g., vibration of the reactor coolant system, the steam generators and the associated containment penetrations) under the condition that the lower vessel head may be submerged in water.
- The probabilities and uncertainties that quantify the potential for a significant water presence in the cavity at the time of vessel failure. This should also include a detailed description of the design features that limit this probability.
- The presence, if any, of drainage paths to and from the cavity, inadvertent system operation, diversion of outflows from various piping breaks and relief valve operations, and the provisions for detection and removal of water from the cavity during operation, with implications on ex-vessel steam explosions.

Please support the discussions of the probabilistic quantification process in terms of available analytic and experimental data.

19-91

Provide the basis for how the issue of over-pressurization of the reactor pit around the time of vessel breach was quantified in the Level-2 PRA. If the issue was decomposed into multiple sub-issues for purposes of quantification, also provide information the decomposition event tree and details of the technical basis for the quantification of the sub-issues.

19-92

In the assessment of HPME-induced DCH, please discuss the technical basis for the values assigned to the input parameters of subcompartment retention fraction and the cavity dispersion fraction. In addition, please discuss in the context of the U.S. EPR and the TCE model what is viewed as the “subcompartment” and what is the likely path or paths for transport of dispersed debris to the upper compartment.

19-93

Please discuss the maximum duration that has been considered in the assessment of long-term containment challenges for the Level-2 PRA. Please provide the technical basis for the selection of this duration and exclusion from consideration of possible later failure of the containment.

19-94

The containment event trees appear to show that, in the event of a large (3 in. or larger) containment isolation failure, or of containment failure resulting from in-vessel steam explosion, most of the remaining top events are not considered, including ex-vessel melt stabilization. For purposes of source term calculation, please clarify if MCCI is assumed to take place or not. If not, please justify how fission product release due to MCCI can be neglected under these conditions.

19-95

Please provide the assessment of the potential for containment failure in the U.S. EPR due to combustion of hydrogen and/or carbon monoxide. Please discuss if and how the presence of carbon monoxide was also treated (for those sequences that may involve protracted MCCI). Consider both deflagration and detonation loads, and the effectiveness of the PARs in limiting the concentrations of combustible gases in the containment. Provide the results from supporting MAAP calculations.

- It is stated in Section 19.1.4.2.1.2 of the FSAR that, in evaluating hydrogen deflagrations for the Level-2 PRA, reference was made to MAAP results in order to determine the amount of hydrogen and oxygen consumed by PARs. Given that the amount of recombined hydrogen is subject to uncertainty (e.g., from uncertainties in timing of release to containment, distribution of gases in the containment, etc.), please discuss the extent by which the results of the deflagration analysis could be impacted by a reduced degree of PAR effectiveness.

- Section 19.1.4.2.1.2 of the FSAR presents conditional probabilities of containment failure due to flame acceleration loads for a number of assessed cases. Please provide details on the formulation of the analysis by which these results were arrived at and the basis for the values of any input parameters used in performing this analysis.

- Please provide the assumed value (or range of values) assigned to the fraction of in-vessel zirconium oxidation, and the technical basis for this (these) value (s).

- In discounting potential containment loads from late hydrogen deflagrations in the Level-2 PRA, the statement is made that the containment would be steam-inerted in this time frame. Please provide the basis for this assumption. Furthermore, a scenario can be envisioned within the containment event progression whereby SAHRS sprays are actuated in the late time frame, which could possibly result in a sudden de-inerting scenario with hydrogen present in the containment. (For example, the CET in Figure 19C-8 includes sequences with initial failure of steam suppression but with later credit to sprays for aerosol removal.) Please explain whether and how such a scenario is treated within the current Level-2 PRA.

- For each PAR unit in the U.S. EPR containment, please provide its model or type and its location (with both room number and room description). Also, for each model or type of PAR present, please provide its nominal hydrogen recombination rate and the influence on recombination rate of pressure, hydrogen concentration, and any other factors. If any equivalent data on recombination rates of carbon monoxide are available, please provide these as well. These data are needed for the planned NRC confirmatory analyses.

19-96

Please provide the following data in support of the NRC severe accident confirmatory analyses:

1. The Annulus Ventilation System conditions that were used in the MAAP 4.07 analyses: • Accident operating pressure,
   • The total volumetric flow rate through both fans,
   • The Fuel and Safeguards Building ventilation exhaust rate, and
   • The elevation of the system with respect to the bottom of the reactor pressure vessel.
2. The Shield Building data that were used in the MAAP 4.07 analyses:
   • Volume and elevation entries,
   • Surface area of the Shield Building, and
   • The aerosol sedimentation area.
3. Diagrams showing the details of the core melt stabilization system (i.e., the reactor cavity, the cavity gate, and the melt discharge channel into the spreading area).
4. The six-group delayed neutron fractions, decay constants, and the neutron generation time applicable to analysis of steam line break inside containment. In addition:
   • The Doppler temperature coefficient (and the associate reference temperature)
   • The moderator temperature coefficient (and the associate reference temperature)
   • The boron coefficient
   • Separately, the total volume concentration of boron inside the chemical and volume control system and the Emergency Boron System (EBS) tanks.
   • The well-mixed boron concentration in the reactor coolant system during the cycle (prior to any accidents).
   • The actuation setpoint and rate of injection from the EBS tank.

Please state the fuel cycle period (e.g., EOL) corresponding to the aforementioned data as applicable.

19-97

Please describe the analytical studies pertaining to the prevention of high-pressure severe accident scenarios, as well as the severe accident uncertainty analyses pertaining to primary circuit depressurization. Include discussions of accident management prior to RCS depressurization, assumptions on PDS valve discharge capacity, and operator response margins. Include the results of the DCH assessment with delayed depressurization, vessel rocketing, and induced RCS rupture, as well as the survivability of depressurization valves under harsh severe accident conditions.

19-98

Please describe the consequences of overpressurizing the reactor pit during blowdown following RPV failure. Include a discussion of potential impacts on severe accident melt stabilization.

19-99

Please describe the base cases and uncertainty analyses of relevant severe accident scenarios that document the calculation matrix of process studies that examine the performance of the U.S. EPR severe accident response features. Explain how the MAAP 4.0.7 code was used in the analyses, and which scenarios and their variations were modeled. Please include, as well, analyses of the main steam line break inside containment scenario, since its cutsets dominate the large release frequency.

19-100

Please provide the analysis that shows how the U.S. EPR containment would maintain its role as a reliable, leak-tight barrier for at least 24 hours following the onset of core damage for the following accident challenges:

- Hydrogen combustion
- Inability to cool core debris
- High pressure melt ejection
- Fuel-coolant interactions

In addition, address equipment survivability under beyond-design-basis accident conditions.

19-101

Please provide a detailed description of the severe accident heat removal system (SAHRS), and a detailed assessment of its performance during representative and bounding severe accident scenarios.

19-102

Provide the bounding environmental qualification conditions (temperature, pressure, humidity, radiation exposure/level) for the instrumentation, controls, and operating components of the SAHRS and the CMSS. Relate these to the calculated severe accident conditions.

19-103

Provide a description and a simplified flow diagram (including valves and orifice plate) for the various in-containment flow paths of the SAHRS in its four modes of operation. Show the location of the piping connections to the flow paths for the active recirculation of water through the core melt spreading area and cooling structure. In addition, please provide the length associated with each pipe section, and the design loss coefficients for all active and/or passive valves for various flow SAHRS paths. Furthermore, please provide the characteristic curve (head vs. flow) for the recirculation pump, the design head and the design pump flow rate.

19-104
Please state the time after initiation of a severe accident within which the SAHRS would have to become available, and relate the consequences were the SAHRS not to be available at this time. Also, please indicate whether there are other possibilities that might be available through SAMG operations to alleviate containment overpressure buildup.

19-105
Please provide the technical basis for debris/melt cooling on the spreading floor (a) in the early time due to gravity driven flow, and (b) over the long-term by pump recirculation. Please provide any experimental data that supports the conditions applicable to the U. S. EPR.

19-106
Please provide a detailed description of the core melt stabilization system (CMSS), and a detailed assessment of its performance during various stages of core melt progression, from vessel failure to long-term melt cooling and retention.

19-107
Please outline the assumptions regarding probabilities of, and containment loads due to, corium quenching following flooding in the spreading area for each distinct containment event tree branch as used for the Level 2 PRA. Include the assumptions and basis regarding base pressure at the time of this phenomenon.

19-108
Please provide the total horizontal surface area available for spreading of molten core debris in the spreading area.

19-109
If available, please provide figures from MAAP calculations showing containment pressurization up to at least 80 hours for (a) a representative sequence with successful quenching and cooling of core debris ex-vessel resulting in continued long-term evaporation of water but without SAHRS sprays; (b) a representative sequence with MCCI but without SAHRS sprays; and (c) a representative sequence with MCCI and with operating SAHRS sprays.

19-110
Please provide the basis for quantification in the Level 2 PRA of the probability that basemat melt-through will occur for branches in the CET involving protracted MCCI. Also, please provide the minimum thickness of the containment basemat assumed for U.S. EPR.

19-111
The CMSS is located in the Seismic Category I Reactor Containment Building, but is not designed for seismic events. Please explain whether this system fails, during a seismic event that initiates a severe accident.

19-112
The core melt plug must be removed from the lower cavity floor to allow for certain shutdown inspection activities. Please state whether it would be a condition for return to power that this plug be satisfactorily back in place.

19-113
Please provide a detailed severe accident equipment survivability evaluation, consistent with directions in SECY-90-016 and SECY-93-087.

19-114
Please provide a detailed description of the combustible gas control system and a detailed analysis of its performance during representative and bounding severe accident scenarios. Include a detailed description of the U.S. EPR hydrogen mitigation strategy.

19-115
For purposes of the source term calculations, please provide and justify the maximum number of broken tubes that were assumed in the case of a temperature-induced SGTR, considering the fact that it is not clear that a single broken tube would necessarily depressurize the primary side of the system to an extent that would mitigate further tube failures.

19-116
Please provide the basis for quantification of and uncertainty with respect to the decontamination factor (DF) assumed in the steam generators for SGTR scenarios with scrubbing available in the Level 2 PRA. If the basis for this quantification involves assumptions regarding the aerosol size distribution, please also provide the assumed distribution and its basis.

19-117
In Table 19.1-20 of the PRA, release category 802 (ISLOCA) shows only about 0.028 maximum release fraction to the environment for aerosols and 0.39 for noble gases. If the radiological release were modeled as being directly to the environment, these figures would seem very low. Please justify the low values of radiological release calculated for this RC. If the reactor building was credited for decontamination, please justify the large DF (approximately 30 inferred), and include the volume, environmental leakage area or characteristics, and deposition or sedimentation surface area inside of the reactor building as used in this MAAP calculation. Please also provide a figure showing the pressure in the reactor building as a function of time as calculated for this scenario, and the basis for any assumptions regarding reactor building failure by overpressurization.

19-118
Table 19.1-20 shows release fractions to the environment calculated for various release categories. Represented by Cs, releases for the containment isolation failure release categories (i.e., RC2xx) are reported to be:

RC201	CIsF with melt retained in-vessel	0.084
RC202	CIsF, RV fails, dry MCCI with sprays	0.022
RC203	CIsF, RV fails, dry MCCI w/o sprays	0.023
RC204	CIsF, RV fails, wet MCCI with sprays	0.019
RC205	CIsF, RV fails, wet MCCI w/o sprays	0.026
RC206	CIsF (2” or less)	0.0082

Please explain why the case of isolation failure with melt retention results in the highest release magnitude (even higher than for vessel failure with dry MCCI and no sprays).

19-119

Please provide the results of the independent peer review of the Level 2 PRA, and a judgment regarding the capability categories of the model in key areas.

19-120

The Large Release classification specification utilized in the FSAR is adapted from guidance listed in Appendix A of NUREG/CR-6595 Rev.1. One limit is that any predicted I, Cs, or Te release fraction above approximately 2.5 to 3 percent is classified as large. At the time of development of that report, the highest licensed MWt for an operating commercial plant was (and is) that for the Palo Verde units at 3990 MWt (NUREG-1350, Vol. 19, Appendix A). The U.S.PWR plant design objective is presented as a rated output of 4614 MWt.

a. Please provide a listing of the equilibrium mid-cycle core radioisotope inventory (e.g., the 60 isotopes normally provided by ORIGEN).
b. Postulating that the NUREG/CR-6595 numbers were intended to provide an early fatality surrogate only for the spectrum of operating reactors and their releases, please provide a discussion of the results that would occur if the limit numbers were linearly scaled to reflect the above differences in plant thermal ratings.

19-121

In 10 CFR 52.47(b)(2) and 10 CFR 51.55 (a), the NRC requires the applicant to prepare an environmental report that includes the cost and benefits of severe accident mitigation design alternatives (SAMDA). The environmental report supporting the SAMDA analysis refers to a Level 3 PRA that was performed to determine the overall risk perspective of the U.S. EPR design. The analysis uses a standard method (template) as provided in the NEI 05-01 for SAMDA analysis in support of license renewal. Review of the methods and assumptions has identified a number of questions related to details of the analyses and conclusions that follow. These include:

a. The report does not provide any details on assumptions and data used in the Level 3 analysis. Please provide the assumptions and data used, including discussion of assumptions related to equilibrium core inventory (considering the potential for high burn up), source term, meteorology, population distribution, evacuation, sheltering, and any other data necessary as part of the input to the MACCS2 program. In addition, please supply the MACCS input that was used to perform the analysis
b. FSAR Figure 19.1-29 provides a range of values for mean and point estimates for internal events core damage frequencies. The core damage frequency used in the SAMDA analysis is 5.3E-07, which is a point estimate value. The mean CDF values given on the same figure has a maximum value 5.1E-06, almost an order of magnitude higher than the point estimate value that was used. Please justify the use of point estimate for the SAMDA analysis, and evaluate the impacts of using the mean CDF on the results of the cost benefit analysis.
c. NUREG/BR-0184 provides a range of doses and costs for occupational exposure, onsite clean up costs, and replacement power costs. The report used only the best estimate values with 3 and 7 per cent discount rates to estimate the range of potential averted costs. Given that these estimates are dated (1992 circa), please elaborate why the analysis did not consider the potential uncertainties in these values. Please evaluate the impact of uncertainties on the results of the cost benefit analysis.
d. In the screening of potential SAMDA, the analysis used the cost estimates from the license renewal submittals, in order to justify the cost effectiveness of implementation of specific SAMDAs. In an operating plant, a backfit may not be cost effective, but this may not be true for the new design. For the new design, the cost would be incremental addition,

whereas, for a backfit is a total cost. Considering these observations, please describe if any one of the items screened out could become cost –beneficial.

e. Another screening criterion refers to "not required for design certification." These are SAMDA items related to procedures and training. Please identify the screened-out list that needs to be considered by COL holders of US EPR design.

f. The basis statements for screening under the "Not Applicable" criterion are not always justified. For example, for SAMDA item "vent MSSVs in containment," the basis for not being applicable is given as "Such a modification would pose design drawbacks, such as increased pressure loading and water inventory within containment, which would exceed the intended benefit of this SAMDA. …." Please provide the impacts in terms of in containment pressure increase and water inventory, and show how these would be different than those of other severe accidents analyzed. Also elaborate on the intended benefit of this SAMDA item for the U.S. EPR design.

19-122

The MAAP 4.07 input deck specifies a total pressurizer free volume of 75 m3, which appears very close to the value that would be estimated from its inside geometry assuming it was completely empty and without accounting for internals such as pressurizer heaters. Please confirm whether pressurizer internals were considered in calculating the pressurizer free volume and, if not, then provide the volume associated with these internals.

19-123

Please provide estimates of the pressure drops and/or form loss coefficients in the reactor core region at normal operating conditions. These should include the values for the lower core support plate/structure, along the fuel rods, and across the upper core plate.

19-124

Please provide examples of several typical MAAP-calculated results for risk-dominant U.S. EPR accident scenarios, including the time evolution of:

- RCS pressure
- Hydrogen produced in-vessel
- Hydrogen mass inside containment (mass and concentration)
- Hydrogen consumed by PARs
- Oxygen (mass and concentration) inside containment
- Steam (mass and concentration) inside containment)
- Other non-condensable gases (mass and concentration) inside containment
- Level of water inside IRWST
- Level of water on the spreading floor, the melt discharge channel, the reactor pit, and other applicable elevations inside containment (showing water drainage into the IRWST)

- Core debris mass inside various containment regions (i.e., reactor pit, discharge channel and the spreading floor)
- Debris penetration distance (axial and radial) in the reactor pit, the discharge channel and the spreading floor
- The rate of release of various fission product groups inside the reactor, and on the containment floor.
- The airborne mass of various fission product groups inside the containment
- The mass of various fission product groups leaking out of the containment
- Total containment pressure (sum of all partial pressures).