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Review of APR+ Level 2 PSA

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Final Report to KINS:

Review of APR+ Level 2 PSA

Revision 2

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1.0 INTRODUCTION AND BACKGROUND

Brookhaven National Laboratory (BNL) assisted the Korea Institute of Nuclear Safety (KINS) in reviewing the Level 2 Probabilistic Safety Assessment (PSA) of the APR+ Advanced Pressurized Water Reactor (PWR) prepared by the Korea Hydro & Nuclear Power Co., Ltd (KHNP) and KEPCO Engineering & Construction Co., Inc. (KEPCO-E&C). The work described in this report involves a review of the APR+ Level 2 PSA submittal [Ref. 1]. The PSA and, therefore, the review is limited to consideration of accidents initiated by internal events. As part of the review process, the review team also developed three sets of Requests for Additional Information (RAIs). These RAIs were provided to KHNP and KEPCO-E&C for their evaluation and response.

This final detailed report documents the review findings for each technical element of the PSA and includes consideration of all of the RAIs made by the reviewers as well as the associated responses. This final report was preceded by an interim report [Ref. 2] that focused on identifying important issues regarding the PSA. In addition, a final meeting on the project was held at BNL on November 21-22, 2011, where BNL and KINS reviewers discussed their preliminary review findings with KHNP and KEPCO-E&C staffs. Additional information obtained during this final meeting was also used to inform the review findings of this final report.

The review focused not only on the robustness of the APR+ design to withstand severe accidents, but also on the capability and acceptability of the Level 2 PSA in terms of level of detail and completeness. The Korean nuclear regulatory authorities will decide whether the PSA is acceptable and the BNL review team is providing its comments for KINS consideration.

Section 2.0 provides the basis for the BNL review. Section 3.0 presents the review of each technical element of the PSA. Conclusions and a summary are presented in Section 4.0. Section 5.0 contains the references.

2.0 REVIEW BASIS: TECHNICAL ELEMENTS OF A LEVEL 2 PROBABILISTIC SAFETY ASSESSMENT

BNL conducted a complete review of the APR+ Level 2 PSA. The review was based on the APR+ PSA Technical Report [Ref. 1], the responses to three sets of RAIs (attached to this report), and the information obtained from KHNP and KEPCO-E&C during an initial meeting on the project, held on July 11-13, 2011, and a final meeting held November 21-22, 2011.

The final review was structured around the technical elements of a Level 2 PSA. The Level 2 technical elements comprise the various portions of analyses that must be carried out in order to obtain the needed results of a Level 2 analysis. Objectives can be defined for each technical element, and these objectives form the basis for what requirements are needed to achieve those objectives. This approach is consistent with that of the ASME/ANS PRA Standard [Ref. 3] for Level 1 and Large Early Release Frequency (LERF) analysis, which has been endorsed (with some exceptions) by the U.S. Nuclear Regulatory Commission (NRC). While no Level 2

PSA Standard is currently available, a draft Level 2 standard is under development. This draft Level 2 PRA standard follows the same format as Ref. 3. It consists of the following: Technical Elements of the Standard, High Level Requirements and Supporting Requirements with the latter divided into three capability categories that reflect the objective of the application for which the analysis is needed and the level of detail required to fulfill that objective.

The Level 2 PSA elements that were used as the basis for this review are consistent with those of the draft Level 2 PRA Standard, and are the following:

1. Level 1-2 Interface
2. Containment Performance Analysis
3. Severe Accident Progression Analysis
4. Probabilistic Treatment of Event Progression
5. Radiological Source Term Analysis

The APR+ concept is in the design stage and, therefore, the design details have not all been worked out as would be the case for an operating reactor; and, of course, there is no operating history. Therefore, it can be expected that data for potential failure modes, likelihood of failure, human error events, etc. are largely based on engineering judgment or rely on experience with reference designs, such as System 80+ and APR 1400, from which the APR+ design has evolved.

3.0 TECHNICAL ELEMENT REVIEW OF APR + PSA LEVEL 2 SUBMITTAL

3.1 Level 1- Level 2 Interface *(Section 3.1 of APR+ Submittal)*

BNL reviewed the plant damage state (PDS) grouping and the carryover of dependencies from the Level 1 analysis to the Level 2 analysis. In particular, the PSA was reviewed for two essential characteristics of the interface between the Level 1 and Level 2 portions:

1. Whether the methodology is clear and consistent with the Level 1 evaluation and creates an adequate transition from Level 1;

2. Whether the interface boundary between the Level 1 PSA and the Level 2 PSA is defined in a manner to preserve the transfer of information (e.g., dependencies) from the Level 1 to the Level 2.

The review confirmed that the interface between the Level 1 PSA and the Level 2 PSA ensured that information from Level 1 PSA (e.g., dependencies, success and failures logic) was properly accounted for, and supplemented as needed, in the Level 2 accident progression analysis. The analysis identified the physical characteristics and accident sequence characteristics at the time of core damage that can influence the severe accident progression.

Accounting for dependencies and incorporating the dependencies in accident logic was implemented by extending the Level 1 event trees in the APR+ PSA. The APR+ PSA submittal stated that the Level 1 event trees were extended to include the status of systems that could influence containment performance. These extended trees were called the PDS event trees. The submittal further stated that the engineered safeguard features, accident mitigation systems, and available documents related to operating guidelines for the APR+ were reviewed to ensure that the Level 1 analyses also considered those systems and operator actions that are important to the containment analyses and accident progression, i.e., the Level 2 analysis. The large number of PDS event tree end states were quantified and propagated through a PDS logic diagram to obtain the frequencies of the PDS bins. No Level 1 sequences were inappropriately screened out in the transition from Level 1 to Level 2. No plant unique issues that could influence the transition from Level 1 to Level 2 were identified.

During the initial review of the PDSs, the BNL review team had difficulty tracing how the output of the Level 1 analysis was transferred to the input of the PDSs. In response to this issue, the team was provided with additional information by KHNP and KEPCO-E&C. This additional information allowed the review team to confirm that the transfer between the Level 1 and Level 2 was appropriate.

The parameters used in the PDS logic diagram to define the PDS bins included the functional status of important systems, variables determined by systems operation (e.g., RCS pressure), the accident initiator type, and the timing of key events. Nine parameters are used in the PSA to define the PDSs. While the BNL review team has concluded that these parameters are generally adequate to define the PDSs, there is an additional parameter that could be significant. This parameter is the status of the secondary side heat removal.

In response to an RAI on why secondary side heat removal was not considered as a PDS parameter, KHNP and KEPCO-E&C provided several reasons as justification. One of the reasons stated was that almost all of the sequences for transient initiators had a failed secondary side heat removal function. This seems to be an anomalous result. For transient and small/very small LOCA initiators in other plant PSAs reviewed, secondary side heat removal does have an impact on the PDS and Level 2 results, so it is difficult to understand why it does not in the case of APR+.

The review team recommends that in a future update of the PSA consideration be given to including this parameter.

3.2 Containment Performance Analysis *(Section 3.2.1 of APR+ Submittal)*

BNL reviewed the severe accident challenges and the failure modes considered for the containment, the structural containment capacity analysis, and the uncertainties considered. In particular, the following areas were reviewed:

(a) All mechanisms of containment failure considered in the assessment of containment capacity.
(b) The method (or methods) to evaluate structural capacity to withstand postulated loads and challenges.
(c) The estimate of the capacity of the containment pressure boundary to withstand loads and challenges generated by core damage accidents.
(d) The character of the uncertainties in the containment failure analysis.

The review of the APR+ Submittal indicated that all failure mechanisms appropriate for a large dry containment type like the one for the APR+ were considered in the Level 2 containment performance analysis. The only mechanism dismissed as negligible was direct core debris impingement on the containment shell, and the reviewers agree with this assessment, given the APR+ containment features and geometry.

The method selected to evaluate structural capacity of the APR+ containment is comparison with the containment of a similar design. The submittal states that at the time of the APR+ Level 2 PSA analysis, plant-specific information was not available for evaluating the APR+ containment design specific ultimate pressure capacity; instead, the ultimate pressure capacity of the containment of the Shin-Kori Nuclear Units 3 & 4 (SKN 3&4) was used. The SKN 3&4 units are APR 1400 designs and their containment is considered sufficiently similar by KHNP and KEPCO-E&C to the proposed APR+ design so that their containment analysis can be a surrogate for the APR+ ultimate pressure capacity evaluation, especially if treated conservatively. The APR+ Submittal does not provide a plant-specific containment capacity analysis. This is one of the many areas, discussed throughout this review, where the APR+ PSA uses a significant amount of information taken from the PSAs performed for other nuclear power plants that are similar to APR+.

In the assessment of containment capacity, there should be consideration of failure mechanisms of the global containment membrane, closure doors, hatches, mechanical penetrations, electrical assemblies, bellows seals, and other major discontinuities in the membrane. The APR+ Submittal does not provide much detail about the various containment features considered in the capacity analysis. The submittal identifies hoop stress failure due to membrane stresses in the cylinder wall of the containment as the likely mode of rupture failure (median pressure capacity of 204.8 psig), while equipment hatch ring plate tearing was identified as the most likely leak failure mode (median pressure capacity of 188.0 psig). The submittal defines leak and rupture in the following manner:

- A leak is defined as a containment breach that would arrest a gradual pressure buildup, but would not result in containment depressurization in less than 2 hours. The typical leak size is evaluated to be of the order of 0.1 ft^2.

- A rupture is defined as a containment breach that would arrest a gradual pressure buildup and would depressurize the containment within 2 hours. The typical rupture size is evaluated to be of the order of approximately 1.0 ft^2.

The fragility curves for the rupture and leak failure modes are presented in the submittal. From the response of two RAIs, the BNL review team understood that for the analysis separate probability curves for the leak and the rupture containment failure modes were calculated and could be combined either in a "leak before rupture" or "rupture before leak" fashion. KHNP and KEPCO-E&C chose the "rupture before leak" option, stating in an RAI response that "The high probability of rupture does not mean rupture occurs in advance of leak. The reason to assume rupture in advance of leak failure is that it causes conservative results in source term release to the environment. The source term with rupture failure mode is more severe than leak failure." While this may not be the most realistic model, the review team agrees that for purposes of an eventual source term calculation, the "rupture before leak" combination would obviously be conservative. The two APR+ containment failure modes are assumed to occur at two different locations, and the independent locations add to the somewhat arbitrary assignment of which mode dominates in what pressure range. The review team believes an approach linking the probability of containment failure mode (i.e., containment failure size) to pressure values, similar to that used in NUREG-1150, would be more desirable.

While there is currently no uniform practice in the US regarding ultimate pressure capacity calculation, the draft Level 2 PRA Standard discusses (for Capability Category II) calculating the ultimate strength of the containment using a validated, computational model that evaluates structural response based on mathematical expressions and correlations that reflect material behavior and governing physical processes and is applicable to the plant specific configuration and conditions. The draft Level 2 Standard cites as an example of ultimate pressure capacity a maximum global membrane strain away from discontinuities of 1 percent for cylindrical reinforced concrete containments. The draft Level 2 Standard also discusses the use of results of experimental measurements of containment performance for specific failure mechanisms. In addition, the US NRC's Regulatory Guide 1.216 provides guidance for containment structural integrity evaluation for internal pressure loadings above design-basis pressure. In particular, the discussion in this Regulatory Guide on acceptable methods for demonstrating that the US NRC's severe accident performance goal is met is instructive, and can provide insights on ultimate pressure capacity calculations for use in PSAs.

As noted above, at the time of the APR+ Level 2 PSA analysis, several calculations related to the APR+ containment performance were not available. Instead, the result of the APR1400 analysis was used considering the similarity and including conservatism. This was the case for analyzing the dynamic capacity of containment, the reactor cavity static strength, the reactor cavity dynamic characteristics, and the response of RCS piping following RV failure. The submittal does not provide a discussion of what the results of the analyses were or what other containment features were examined in the capacity analysis.

The sources of modeling uncertainty or assumptions used in the deterministic analysis of containment failure were not identified or discussed in the submittal. Therefore, there was also no characterization of how the containment strength or resistance to failure might be affected by modeling uncertainties or assumptions.

The review of the APR+ Level 2 containment capacity analysis concluded that on the whole the APR+ containment appears robust and able to deal with many severe accident challenges. The appeal to a reference plant containment analysis rather than a plant-specific analysis is understandable given that the APR+ is still in the design stage. However, more discussion of the failure mechanisms analyzed for various containment features, as well as a discussion of the uncertainties and assumptions that can have a significant impact on the results, would still be desirable. Presumably, as the APR+ design is finalized, these discussions, along with APR+ containment-specific ultimate capacity calculations that provide more information on material properties assumed, response to increased temperatures as well as pressures, and justify failure locations and size, will be carried out.

3.3 Severe Accident Progression Analysis *(Sections 3.2.2 and 3.2.3. of APR+ Submittal)*

BNL reviewed the bases used to generate quantitative estimates of containment pressure and temperature, combustible gas concentrations, and other loading mechanisms. The review covered the assumptions made and uncertainties considered. In particular:

(a) The review considered if the deterministic (computational) model for generating each of the quantitative estimates of the parameters is appropriate, given the quantitative metrics of severe accident behavior that the deterministic analysis is supposed to calculate.

(b) The calculations were reviewed to determine if they appropriately:

- chronologically order, and quantitatively estimate the time of, important events;
- calculate the plant response conditions that could challenge the ultimate capacity of the containment;
- develop success criteria for the mitigating systems used in the Level 2 PSA;
- develop success criteria for operator actions credited in Level 2 PSA;
- generate quantitative measures of parameters used to estimate probabilities of uncertain events and phenomena; and
- characterize fission product releases, including timing and location of the release within containment.

(c) The review assessed whether the effects of uncertainties in calculating boundary conditions, severe accident phenomena and plant response to environmental conditions beyond the design basis of the plant were properly accounted for.

The APR+ Level 2 PSA analyses used a general containment event tree (CET) and several special CETs, all enhanced by decomposition event trees (DETs) to chronologically order and structure the accident progression events. A step in the quantification of these event trees is the use of deterministic analyses to generate quantitative measures of parameters that are then used to estimate the probabilities of uncertain events and phenomena in the trees. It is these deterministic calculations that are the focus of the review of this PSA element. Usually a computer code, such as MAAP or MELCOR, run with plant-specific inputs, is used to carry out these calculations.

However, the BNL team found that APR+ plant-specific calculations were not carried out. The APR+ PSA Submittal states that: "For the quantification of DET branch probability, APR 1400 and SKN 34 PSA and severe accident analysis results were considered first because APR+ plant specific severe accident analysis result is not available at this time. If the results for APR 1400 and SKN 34 specific analysis are not available, other applicable analysis and/or experiment results were used for quantification." The submittal goes on to acknowledge that: "…caution should be taken in directly or indirectly applying information generated for other plants, i.e., some modifications are probably necessary because of plant specific considerations."

The review concluded that while the CETs and DETs are for the most part comprehensive and acceptable for tracking severe accident progression, since the APR+ PSA uses almost exclusively information taken from reference plants that are considered to be similar to APR+, the usefulness of the PSA is somewhat limited. If the BNL reviewers were to assign a Capability Category, as used in the ASME/ANS PRA Standard for Level 1 and LERF PSA calculations, this would be Capability Category I for the current APR+ Level 2 PSA. Also, the APR+ Submittal does very little to justify the applicability of the reference plant calculations or their results to the APR+ severe accident progression. While general statements about plant similarities are made in the submittal, there are no discussions on how quantitative results of reference plants calculations were adapted to the scenarios of interest in the APR+.

An example of where additional plant-specific analysis would enhance the level of detail and completeness of the PSA is the treatment of bypass sequences. In the APR+ Level 2 PSA, the separate branches for accident sequences, such as steam generator tube rupture (SGTR) and interfacing systems LOCA (ISLOCA), that bypass the containment building were not further developed. This is a conservative assumption because, depending on the flow path of the radioactive material from the damaged core to the environment, significant scrubbing of the release can occur if any element of the flow path of the release is submerged in water. As scrubbing can potentially significantly reduce the quantity of radioactive materials released, this phenomenon should be considered in the formulation of the PDSs, the containment accident progression and source term categorization for bypass sequences. For example, SGTR accident sequences in which the secondary side of the steam generators may or can be flooded should be analyzed separately. Similarly, in the case of ISLOCA, the possibility that the safety injection lines where the break occurs may be in a submerged location or that a portion of the release path is through a pool of water needs to be investigated. If it is unlikely that break locations will be submerged, because of plant geometry, for example, then this should be documented in the submittal. (During the review meetings, the BNL team learned that it is the intent of the APR+ designers to have floor drains installed throughout the plant so it is not likely that any release path of an ISLOCA will be submerged). In the current PSA, mechanistic plant specific source terms for the various containment failure modes have not been calculated. However, based on the submittal and a number of RAI responses, it seems that the intent is to eventually perform mechanistic calculations using the MAAP code to provide estimates of the quantities and timing of radionuclides released. The review team recommends, therefore, that consideration of scrubbing of the releases by any overlying pools of water be included when performing the mechanistic calculations.

Regarding the use of reference plant information, the review team also found that sometimes the information used from reference plants does not reflect the most up-to-date research findings on particular severe accident issues. An example where the use of results from similar nuclear power plants may not contain up-to-date information is the quantification of RCSFAIL branches in the decomposition event tree RCSFAIL DET ("Mode of thermally induced upper RCS pressure boundary failure"). The quantification of the APR+ Level 2 PSA was based on data drawn from the System 80+ PSA, NUREG-1150 [Ref. 4], NUREG-4550 [Ref. 5] and APR 1400 PSA studies. The branching probabilities of the decomposition event tree for high RCS pressure sequences were allocated as follows:

1.	No RCS failure:	P(NORCSFAIL)	= 0.632
2.	Hot leg failure:	P(HLFAIL)	= 0.35 (CESSAR)
3.	SGTR:	P(SGTR)	= 0.018 (NUREG-1150)

After the NUREG-1150 study was completed, a number of other studies, including NUREG/CR-6075 [Ref. 6] and NUREG-1570 [Ref. 7], were carried out to further understand risk-significant severe accident phenomena. These studies addressed direct containment heating and severe accident-induced RCS boundary failure. Two phenomena that are highlighted in these studies can be important in the Level 2 PSA for the following reasons:

1. If hot-leg/surge line failure occurs before vessel breach, the RCS pressure will be decreased significantly. This will change high pressure accident sequences to low pressure sequences, impact the possibility of in-vessel injection, and hence affect the likelihood of subsequent events, such as dynamic containment failure and early containment failure. Eventually, this possibility can substantially impact the overall risk insights of the assessment.

2. Although temperature-induced hot-leg or surge line failure will, in general, lead to less consequential source terms and thus contribute to decreased offsite consequences, temperature induced SGTR (TI-SGTR) can have the opposite effect by leading to larger source terms.

Therefore, the phenomenon of temperature-induced failure of the RCS boundary that affects the source terms which can be released to the environment needs to be considered in the Level 2 PSA. It is, therefore, recommended that the results of NUREG-1570 and NUREG/CR-6075 should be consulted in the APR+ Level 2 PSA.

In many cases, the APR+ PSA Submittal, as well as some RAI responses, justified the use of reference plant information and lack of some severe accident development as acceptable due to conservatism and simplicity of the analyses. However, in general, it is not the objective of a PSA to provide a conservative result; rather it should provide realistic results. PSAs that are biased to conservatism have limited use.

The most noticeable omission of severe accident progression calculations is the omission of representative source term calculations. Such calculations are usually found in a Level 2 PSA, but not the APR+ Level 2 PSA. Neither adapted reference plant nor APR+ specific source term estimates were provided. This is discussed further in Section 3.5, Radiological Source Term Analysis, below.

As noted above for containment performance, the use of reference plant information rather than a plant-specific analysis for the severe accident progression analysis appears acceptable, given that the APR+ is still in the design stage. However, as the APR+ design is finalized, more plant-specific calculations and/or calculations justifying adaptation of reference plant information need to be provided for determining the order and timing of severe accident events, and the phenomenological parameters of interest. A realistic modeling tool should be used and its technical basis provided. The reasonableness of results should be demonstrated.

Again, the sources of modeling uncertainty or assumptions used in the deterministic accident progression analysis were not identified or discussed in the submittal. Therefore, there was also no discussion of how the accident progression analysis results might be affected by the modeling uncertainties or assumptions.

3.4 Probabilistic Treatment of Event Progression *(Section 3.2.3 of APR+ Submittal)*

BNL reviewed the complete structure of the framework used to represent credible severe accident progressions. BNL also reviewed the probabilistic assessment of the accident phenomena, as well as the modeled equipment and human actions. In particular, the review focused on whether:

(a) The methodology is clear and consistently linked with the Level 1 evaluation to create an adequate transition from Level 1;
(b) Operator/Technical Support Center (TSC) Human actions, mitigation systems and phenomenological behaviors that can alter the event progression are adequately considered and characterized;
(c) Dependencies are appropriately reflected in the model structure;
(d) Phenomenology is appropriately characterized and modeled;
(e) Analyses are provided to support equipment success criteria, time windows for operator human action, access requirements for operator human actions, and other recoveries;
(f) End states are defined in sufficient detail so that these can be characterized in terms of release timing, containment failure mode, release distribution and magnitude; and
(g) The frequency of the severe accident sequences leading to the defined end states is appropriately determined.

Regarding item (a) above, the methodology used in the framework for representing credible accident progressions is clear and consistent with the Level 1 evaluation. As noted in Section 3.1, the APR+ analysts used extended Level 1 event trees, called PDS event trees, to transfer the necessary information from the Level 1 analysis to the Level 2 analysis. The extended Level 1 sequences were binned into PDSs according to certain characteristics, and the accident sequences were propagated via containment event trees (CETs), enhanced by decomposition event trees (DETs), to chronologically order and structure the accident progression events. This approach appropriately addresses the requirements of (a) above and also is useful for appropriately reflecting dependencies in the model structure, as required by item (c) above.

This approach also partially addresses item (b) but not entirely. The APR+ methodology for the most part appropriately considers the phenomena and human actions that can alter the accident progression, but sometimes additional development of a potential scenario would be helpful. An example is the exploration of the scrubbing of releases associated with some bypass sequences, as described in Section 3.3 above.

The characterization and modeling of phenomena listed in item (d) was somewhat superficial in the APR+ Level 2 analysis, as already partially discussed in Section 3.3 above. The APR+ submittal relies almost exclusively on reference plant analyses for characterizing the phenomena of interest, and estimating their branch point probabilities in the CET and DETs. The analysts' degree of belief of the assigned branch point probabilities is based on engineering judgment that arises largely from these other reference plant analyses. The submittal makes the following statement: "Another way of obtaining the branch point probabilities is to use existing and available models for the phenomena and processes represented by this event (e.g., the MAAP4 code). The important parameters affecting this event should be identified through sensitivity analyses. Detailed sensitivity analyses using deterministic models and the application of engineering judgment will aid the quantification of the CET." However, no plant-specific analyses were carried out for the potential phenomena in the APR+ accident progression. Much of the quantification is carried out by using a likelihood scale that assigns a probability range and a nominal value to qualitative verbal descriptors, such as "Certain, Highly Likely, Very Likely, Likely, Indeterminate, Unlikely, Very Unlikely, and Highly Unlikely." However, the actual values assigned to the branch point probabilities are in many cases more conservative (i.e., biased toward a less favorable outcome for the event) than the likelihood scale would suggest.

The engineering judgment used by KHNP and KEPCO-E&C to quantify the DETs and CETs seems generally reasonable but, in some cases, would be enhanced by additional clarification, which could often be provided by plant-specific deterministic code calculations that are currently lacking. Additional plant-specific calculations would improve the level of detail and completeness of the PSA.

Regarding item (e), the APR+ Submittal is sparse on detail on analyses supporting equipment success, time available for human action, and other recoveries. An example of where more consideration is warranted is provided by the discussion in the submittal on the quantification of the ECSBS unavailability in the MELTSTOP and CS-LATE DETs. To determine the availability of the emergency containment spray backup system (ECSBS) (which requires operator action) in these DETs, reference is made to the Level 1 PSA reliability data. The ECSBS unavailability in the Level 1 PSA was estimated to be 5.75E-2 including a human failure event (HFE) probability of 1.45E-2. A detailed analysis with consideration of the harsh environment anticipated after the onset of core damage was not performed for the procedures. Instead, to consider the effect of the harsh conditions and high stress due to the threat of core damage, the ECSBS unavailability (reflected by the parameter CSRECSBS) in the Level 2 analysis was changed from 5.75E-2 to 0.1, which translates to an increase in the human failure event probability of less than a factor of 2 also. The submittal states that this was done to “account for uncertainty and maintain conservatism.” In an RAI, the review team requested more justification for this value and asked for a sensitivity analysis of the effect of the unavailability of ECSBS. The RAI response included a sensitivity calculation where the parameter CSRECSBS was set to 1.0. This indicated that the late containment failure probability was quite sensitive to the ECSBS unavailability, increasing from 5% to 45% of all outcomes when CSRECSBS increased from 0.1 to 1.0. Therefore, the review team suggested that a sensitivity calculation of late containment failure assuming a HFE with a probability of about 0.1, i.e., a factor of about 6 higher than estimated in the original Level 1 calculation, be used in the Level 2 calculation. This would make the CSRECSBS unavailability approximately 0.2 and is likely to lead to significant increase in the late containment failure contribution. The review team believes the value of 0.1 for the HFE is not overly conservative, and a sensitivity analysis using this or a comparable value would be useful. Because this value is so important for the overall results, additional effort to justify the modeling and data should be applied.

With respect to item (f), end states are well defined in terms of the events in the accident progression, but since no source term calculations were performed, their definition is incomplete because release timing and magnitude is not available. This is further discussed in Section 3.5 below.

The framework for the event progression does lend itself to appropriately determine the frequency of the severe accident sequences, as required in (g) above.

3.5 Radiological Source Term Analysis *(Section 3.4 of APR+ Submittal)*

BNL reviewed the radionuclide release categories arrived at and their characteristics. BNL also review the quantification of the radionuclide release categories, including the uncertainties considered in the radionuclide release and transport. In particular the BNL review considered:

(a) The Radionuclide Release Categories (source term bins) arrived at in the PSA and their characteristics (e.g., release fraction or magnitude, timing of release, etc.);

(b) The method used for determining the source term for each category of Level 2 accident sequences;
(c) The calculations performed to quantitatively characterize source terms; and
(d) The characterization of uncertainties considered in the radionuclide release/transport phenomena.

One of the end products of a Level 2 PSA involves the characterization of the release from the containment to the environment. Usually the characterization involves an analysis that provides a description of the type, quantity, radioactive material species, chemical form, thermal energy, and timing of the release of radioactive material from the plant during an accident sequence, i.e., the source term. However, the metrics used to define a source term can vary, depending on the objective and intended application of the PSA. For example, if the Level 2 PSA results are to be used in a Level 3 consequence assessment, it may be necessary to provide more detailed source term information than if no Level 3 assessment is to be performed, as is the case for the APR+ analysis under review.

In practice, release categories, which consist of a group of radioactive material releases expected to result in similar consequences, are defined. This grouping is based on common attributes, such as common initiating events, combination of successful and failed safety functions, release magnitude, release timing and location, and radioactive material species that are released from the plant as a result of an accident. The release categories are then each characterized by a bounding mechanistic source term.

The APR+ Level 2 PSA correctly defines the source term release categories (STCs) as the grouping of containment event sequences that have similar characteristics from the standpoint of fission product release. Section 3.3 of the PSA submittal identifies the following steps that need to be followed for analyzing the STCs:

1. Define the source term grouping parameters,
2. Develop the source term category logic diagram,
3. Quantify the logic diagram based on the results of the containment event tree quantification, and
4. Perform deterministic code calculations for selected representative sequences in each STC to obtain magnitude, timing, energy, etc. of radionuclide release to be associated with each release category.

The grouping parameters were defined as follows: (1) containment bypass, (2) containment isolation, (3) in-vessel melt retention, (4) time of containment failure, (5) mode of containment failure, (6) containment spray system status, and (7) debris coolability. The BNL review team agrees that these are adequate to group the accident progression sequences into STCs for further analysis.

However, while the APR+ Level 2 PSA has carried out Steps (1) through (3) above, Step (4), deterministic code calculations, has not yet been performed; hence, there are no plant-specific mechanistic source term analyses. The team recognizes that the intent is to eventually perform mechanistic calculations to provide estimates of the quantities of radionuclides released to the environment.

The current source term analysis in the PSA is, therefore, limited to determining the characteristics (e.g., containment bypassed, early containment failure, etc.) of the releases and their corresponding frequency. This approach can identify which release classes are the most frequent, and, to some extent, which are likely to be the most risk significant. However, the other characteristics of the radionuclide release to the environment, such as type, quantity, radioactive material species, chemical form, or thermal energy, have not been calculated at this time.

Therefore, when comparing the APR+ source term analysis to the review items (a) through (d) above, the BNL review team found that (a) and (b) have been satisfactorily carried out, but (c) and (d) have not. The calculations to quantitatively characterize the source terms were not carried out. As with the other PSA elements, the sources of modeling uncertainty or assumptions used in the source term analysis were not identified or discussed in the submittal.

The lack of quantified source terms leaves the APR+ Level 2 PSA incomplete and has implications for the objectives stated in the submittal. In the Objectives and Scope section (Section 1.1), the submittal states that an objective of the PSA is to demonstrate that: “Cumulative frequency should be less than 1.0E-7 per reactor year for sequences resulting in greater than 0.01Sv for 24 hours at the site boundary.”

Since the quantities of radionuclides released in various accidents and the timing of the releases have not yet been calculated, this objective cannot be demonstrated to have been met. This issue was raised in an RAI. The review team suggests that this objective be removed until deterministic source terms are calculated. These source terms can then be combined with an off-site consequence model to calculate the dose at the site boundary for comparison with the design objective. As an alternate approach to the dose at the site boundary, a conservative estimate of LERF can be calculated by summing the frequencies of bypass and early failure events. However, LERF calculations cannot be used to justify an estimate of frequencies for relatively low doses at the site boundary, such as 0.01 Sv, over long time periods of exposure, i.e., over a period of 24 hours.

An argument that the stated design objective of “cumulative frequency less than 1.0E-7 per reactor year for sequences resulting in greater than 0.01Sv for 24 hours at the site boundary” may be met without source term calculations is made in one of the RAI responses. The response states that “the design objective related to dose at the site boundary is expected to be met in the APR+ design because the estimated containment failure frequency or LRF, 9.15E-08/yr, is less than the target value.” However, this frequency is really at the threshold of the 1.0E-7 per reactor year criterion and since in the present PSA only internal events are considered, it is likely to increase significantly when other, i.e., external, hazards are accounted for. Even with only internal events, an increase in a parameter like the human error event probability, discussed in the example in Section 3.4 above, would likely send the cumulative containment failure frequency above the 1.0E-7 per reactor year threshold. This is another reason to drop this design objective, at least for now, until quantitative source terms have been estimated that would allow site boundary doses to be computed.

4.0 CONCLUSIONS AND SUMMARY

In this section, the review team presents its views on the capability or 'acceptability' of the APR+ Level 2 PSA. Whether the PSA is acceptable is of course strictly up to the Korean nuclear regulatory authorities, and the review team is providing its comments for KINS consideration.

PSAs are generally used to support a wide range of risk-informed applications that require a corresponding range of PSA capabilities. It has long been recognized that the acceptability of a PSA for a particular application depends on the specific needs of that application. This concept is reflected, for example, in the establishment of capability categories in the ASME/ANS PRA Level 1/LERF Standard, an idea that is also being continued in the various other PSA Standards being drafted, including the draft Level 2 Standard. In these standards, PSA capability is broadly defined in terms of scope and level of detail, plant specificity, and realism. For a Level 2 analysis, scope and level of detail refers to the degree to which the analysis sufficiently captures the important physical phenomena relevant to the plant design. Plant specificity refers to the degree to which plant-specific information is incorporated such that the as-built and as-operated plant is addressed. Realism is the degree to which a realistic treatment of events and phenomena is incorporated such that the expected responses of the plant and containment are addressed.

As a plant moves through the design stage to construction, licensing and operation, the scope and level of detail, plant specificity, and realism of PSAs conducted during these stages would be expected to increase. While the level of conservatism in the PSA would in general be expected to decrease during these stages, as more detail and more realism are introduced into the analysis, this should not be assumed to be universally true[1].

In conducting their review of the APR+ Level 2 PSA, the BNL review team kept in mind that the analysis described in the PSA submittal was carried out for a plant in the design stage, at a point where a number of the details of the design were not yet finalized. The team recognized that many more assumptions need to be made in this design stage PSA than in a PSA for a constructed and operating plant. Therefore, the conclusions of the team on the APR+ Level 2 PSA are presented primarily regarding the submitted PSA as a design stage PSA. At the end of

[1] The treatment of hydrogen distribution and combustion within a large dry containment might provide a counter example. By assuming the complete combustion of a hydrogen mass representing oxidation of 100% of the Zircaloy cladding in the core to be uniformly distributed within the containment free volume, the resulting pressure increment might be very small. However, if a more refined spatial treatment of hydrogen transport and mixing within the containment is considered, very high concentrations might be estimated in small local regions of the containment which, if ignited, could threaten containment integrity.

this section, the review team also presents some brief comments on what may be needed in an updated PSA when the design is finalized.

For an initial design stage PSA, the review team believes most portions of the APR+ Level 2 PSA are nominally acceptable. For a design stage PSA, the scope and level of detail should have a resolution and specificity sufficient to identify the initiating events, system failures, system operating characteristics, mechanisms of containment failure and severe accident progression phenomena that contribute to the significant accident progression sequences and thus the significant release categories that are the result of the analysis. In terms of plant specificity, in the design stage the use of generic data/ models is acceptable except for the need to account for the unique design and operational features of the plant. As for realism, in the design stage bounding or conservative characterization of the frequency and physical characteristics (magnitude, timing, etc.) of radiological releases for the accident sequences generated in the Level 2 PRA should be sufficient, unless more data and information is available that would justify a less conservative approach.

The APR+ Level 2 PSA reviewed has met the above stipulations in many areas:

The PSA is generally well constructed and follows established approaches for performing a Level 2 PSA. The transfer of information between the Level 1 and Level 2 portions of the PSA was found to be appropriate. The containment event trees and associated decomposition event trees are for the most part comprehensive and acceptable for tracking severe accident progression. As a whole, the APR+ containment appears robust and able to deal with many severe accident challenges.

However, the review team feels that the PSA is incomplete in its current form. The PSA appears incomplete in three main areas:

1. Source term estimates have not been provided. Mechanistic code calculations to assign source terms to the release classes have not been carried out. Neither has any source terms from reference plants been discussed. This is a significant omission for many reasons, including the fact that the design objectives stated in the submittal cannot be demonstrated to be met.

2. Sufficient justification of applicability of reference plant data has not been provided. The PSA uses a significant amount of information taken from the PSAs performed for other plants that are similar to APR+. As discussed above, this is certainly acceptable and even inevitable for a design stage PSA; however, use of reference plant information and analyses need to be justified as applicable to the plant and accident conditions to which they are applied. This kind of justification is very sparse in the APR+ Level 2 PSA Submittal. While reference plants analyses are often referred to, the specifics of why they are applicable to the accident scenario or phenomena under discussion are usually not provided. If modifications or adaptations were made to the reference plant information to make it more applicable to the APR+ this is also not mentioned. In

addition, sometimes the information from reference plants does not reflect the most up to date research findings on particular severe accident issues. Since the reference plant information appears as the main basis for the engineering judgment used to quantify the DETs and CETs, additional justification for its applicability, which is currently lacking, should be provided.

3. There is no comprehensive identification and characterization of the important uncertainties and assumptions of the Level 2 analysis. For each of the Level 2 PSA elements discussed in Section 3 above, the important sources of modeling uncertainty and assumptions should be identified. This also includes assumptions used to adapt deterministic calculations from a reference plant. This listing of uncertainties and assumptions should be carried out for of the containment failure analysis, the deterministic accident progression analysis, the probabilistic accident progression, and the source term analysis. For each identified source of model uncertainty and related assumption its effect on the corresponding element results, i.e., containment capacity, accident progression, etc. should also be estimated. For example, once the sources of model uncertainty and assumptions used in the deterministic analysis of containment failure are identified, the effect on containment strength of each uncertainty or assumptions should be characterized. Similarly, for each identified uncertainty and assumption in the accident progression analysis the effect on the accident progression analysis should be characterized, and so on. While the APR+ Level 2 PSA Submittal does provide four sensitivity analyses, substantially more should be provided. At the design stage it is very important to highlight where the critical assumptions and model uncertainties are embedded in the analysis. Additional discussions on model uncertainty and assumptions, along with more sensitivity calculations are needed.

The concerns above broadly reflect the conclusions on specific items reached at the final meeting between BNL, KINS, KEPCO-E&C and KHNP. These conclusions were:

- For the design stage PSA more generally accepted Level 2 PSA criteria, like conditional containment failure probability, or large release frequency, should be used instead of the release criteria cited in the submittal.
- Some more detail should be explored for some of the accident sequences, such as the bypass sequences.
- Post NUREG-1150 documents with more up to date information on thermally induced reactor coolant system boundary failure modes should be consulted for updating some of the split fractions. This includes further investigation of the consequences of hot leg failure in terms of allowing potential in-vessel coolability of the damaged core and the potential effect on the accident progression.
- More investigation of the human error event probability affecting ECSBS unavailability in the MELTSTOP and CS-LATE DETs is needed to justify the value used for the base case.
- Additional sensitivity analyses should be performed, including some where several parameters are varied at the same time.

When the APR+ design is finalized, presumably an updated, more comprehensive PSA that builds on the design stage PSA will be carried out. For this stage, in addition to satisfying the issues raised above, the reviewers believe that significantly more plant-specific calculations with respect to phenomena, human actions, and containment performance are warranted.

In terms of scope and level of detail the resolution and specificity of the PSA should be sufficient to identify the operating modes, initiating events, system failures, system operating characteristics, mechanisms of containment failure and severe accident progression phenomena that contribute to all identified accident progression sequences. The plant-specificity should be upgraded so that plant-specific models are used for the containment capability calculations (e.g., use of a finite element model), the significant containment integrity challenges and the significant fission product release characteristics. In terms of realism, the characterization of the frequency and physical characteristics (magnitude, timing, etc.) of radiological releases for significant accident sequences generated in the Level 2 PRA should be realistic.

In addition the uncertainty analysis should also be still more comprehensive. For each assumption and source of model uncertainty identified throughout the PSA, how the PSA model is affected should also be identified (e.g., introduction of a new basic event; changes to basic event probabilities; change in success criterion; changes to radionuclide release frequency, magnitude, or timing; introduction of a new initiating event). Besides identifying and characterizing the sources of model uncertainties and assumptions, the uncertainty range for branching probabilities (split fractions) should be characterized to permit a quantitative study using structured sensitivity analysis. The radiological source term analysis should also estimate the effects of uncertainties including an estimate of the uncertainty interval for the frequency of each release category.

5.0 REFERENCES

[1] APR+ Technical Report (Probabilistic Safety Assessment), KHNP, June 2011.

[2] Important Issues Resulting from the APR+ Level 2 PSA Review, Preliminary Report, BNL Letter Report, November 2011.

[3] ASME/ANS RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

[4] NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, December 1990.

[5] R.C. Bertuccio et al., “Analysis of Core Damage Frequency: Surry Unit 1,” Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Rev. 1, SAND86-2084, April 1990.

[6] Pilch et al., “The Probability of Containment Failure by Direct Containment Heating in Zion,” NUREG/CR-6075, SAND93-1535, Sandia National Laboratories, Albuquerque, NM, 1994.

[7] NUREG- 1570, “Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture,” U.S. Nuclear Regulatory Commission, Washington, DC, 1998.

Attachments:

Requests for Additional Information and Responses

1st Round RAIs and responses

2nd Round RAIs and responses

3rd Round RAIs and responses

Response to BNL RAIs for Level 2 PSA of APR+ (Round 1)

1. Justification and use of the table to quantify analyst's degree of belief:

At the end of Section 3.2.3.1 of the report there is a table which assigns ranges and nominal values to descriptors used in the text describing relative likelihoods of events quantified in the DETs. The nominal values used in the table are often not near the midpoint of the range. What rationale was used for selecting the nominal values? In addition, the descriptors and nominal values used in the quantification of the DETs are not always consistent with the descriptors and the nominal values assigned in the table. Please clarify.

Response to RAI 1:

Nominal values in the table indicate nominal values for the verbal (or linguistic) probability but don't indicate mean probability from a distribution. We used the nominal value as a representative value of the likelihood range when branch probability in DET was assigned based on engineering judgement. This concept of verbal (or linguistic) probability was incorporated into the Level 2 PRA CET quantification by KAERI(Korea Atomic Energy Research Institute) with the help of NUS. This quantification concept using verbal probability was reviewed by KINS as a part of PRA review and it has become a Korean Level 2 PRA practice.

One of the reasons for the inconsistency between the assigned probability in DET and descriptor (description) is that a philosophy to apply probability in APR+ Level 2 PRA CET quantification. Basically we want to introduce some conservatism for phenomena with higher uncertainty so that our base model has some conservatism if the impact of the conservatism on final results is not significant.

If there is any other inconsistency between the assigned probability in DET and descriptor (description), the probability is correct. There might be inconsistencies between the assigned probability in DET and descriptor (description) due to poor English language proficiency.

2. Background of thermally induced upper RCS failure probability:

The quantification of Event 3 (RCSFAIL) of the DET for *Mode of Thermally Induced Upper RCS Pressure Boundary Failure* relies on the System 80+ PSA, NUREG-1150, NUREG/CR-4550, and the APR 1400 PSA. Were any of the studies carried out subsequent to NUREG-1150, such as NUREG/CR-6109, "The Probability of Containment Failure by Direct Containment Heating in Surry," about hot leg failure and the likelihood of DCH, consulted? Was the information in NUREG-1570 on SGTR consulted for quantification?

Response to RAI 2:

Neither NUREG/CR-6109 nor NUREG-1570 have been explicitly consulted for the quantification of DET RCSFAIL.

Although the probability for Mode of Thermally Induced Upper RCS Pressure Boundary Failure was based on NUREG/CR-4551 (for Surry), we modified the probability to incorporate conservatism to consider uncertainty because higher probability for this phenomenon will lead to optimistic result.

3. **Clarification on MAAP4 calculation used to support MELTSTOP quantification:**

The discussion of the two MAAP4 calculations carried out to support the quantification of the DET for *Core Melt Arrest* (CET Event: MELTSTOP) is somewhat confusing. Please clarify the following two paragraphs below, reproduced from the report:

“Two MAAP4 calculations for the reference plant have been used to support this DET. The first one is for the successful safety injection sequence under core damage. The result shows the reactor vessel does not fail even if the support plate fails and the core debris exists in lower head of reactor vessel. The other one is for the case of containment failure before reactor vessel failed. The MAAP4 calculation shows that without containment heat removal, cooling core debris results in containment failure before the reactor vessel failure. While MAAP4 calculations show core melt arrested, the phenomena involved in this event is still uncertain as mentioned above.

Energy transferred from core debris to water overlying the debris in the reactor vessel, heats up and vaporizes water. The resulting steam is returned to the water pool in the containment as water via containment sprays if they are available. Consequently, produced steam is not a threat to the containment. In the event that the containment heat removal function (CHRCOOL or CSRECSBS in DET) is lost, containment overpressurization will ultimately result and core melt will occur because the vessel will breach.”

Please explain sequences 7 and 8 on the MELTSTOP DET.

Response to RAI 3:

The reference plant MAAP analysis can be briefly clarified as follows. The analysis took a credit for gap cooling model of reactor vessel lower head in the MAAP.

Case 1: Success of in-vessel injection AND success of containment heat removal
- ⇨ Core melting stops in vessel and containment integrity is maintained

Case 2: Success of in-vessel injection AND failure of containment heat removal
- ⇨ Core melting stops in vessel and containment integrity is not maintained
- ⇨ Failure of ESF pumps due to lack of NPSH after containment failure
- ⇨ In-Vessel injection failure
- ⇨ Core melt progresses and leads to vessel failure

Case 3,4: Failure of in-vessel injection
- ⇨ Core melting process proceeds to vessel failure before containment failure

For the probability assignment in the DET MELTSTOP refer to the table below;

Case No	In-vessel Injection Status	Containment Spray Status	Probability for Vessel Failure before containment Failure	Probability for Containment Failure Before Vessel Failure
Case 1	Yes	Yes	0.05	N/A
Case 2	Yes	No	0.05	0.95
Case 3	No	Yes	1.00	N/A
Case 4	No	No	1.00	N/A

Sequences 7 and 8 of the MELTSTOP DET correspond to Case 2 in the table above.

4. Probability of non-recoverable failure of containment sprays:

During the discussion of the quantification of Event 5 CSRECSBS: *ECSBS Operation of the DET for Core Melt Arrest (CET Event: MELTSTOP)* the report states:" Because the conditions after vessel breach are very harsh and with high stress on operators, the probability of non-recoverable failure of containment sprays is conservatively assumed to be 0.1." Please explain further why 0.1 would be considered a conservative value, given the harsh conditions outlined.

Response to RAI 4:

Level 1 internal event PRA estimated ECSBS unavailability including Human Failure Event. According to the result, ECSBS unavailability is 5.75E-2 which included a human failure event probability of 1.45E-2. The ECSBS unavailability is based on the condition prior to core damage. To consider the harsh condition and high stress due to threat of core damage, we modified the ECSBS unavailability as 0.1 for uncertainty consideration and maintaining conservatism.

5. P(leak-only) in Table 3.2-1 (Containment Fragility curve):

The probabilities for each mode of containment failure as a function of pressure are shown in Table 3.2-1 of the report. Please explain the meaning of the P(leak-only) column. The relationship of the other columns seem self-evident from the probabilities, i.e., P(Rup) +P(Leak) = P(Damage), and P(Intact) = 1.0 - P(Damage), but what is the significance of P(leak-only)?

Response to RAI 5:

P(leak-only) is the leak-type containment failure probability independently evaluated from P(Rup). P(Leak) is the leak-type containment failure probability provided the containment integrity is maintained against rupture-type containment failure due to containment pressure load. P(leak) can be calculated as P(leak-only)*(1-P(Rup)) on a cumulative distribution basis.

As you mentioned in the question, P(leak-only) is an intermediate result to get final containment fragility curve. In the Level 2 PRA, P(leak-only), itself, is not directly used for the containment failure frequency quantification.

6. HMS Success criteria (PAR, Igniters):

In Section 3.1.1.9 the HMS is discussed in the report. The statement is made that

"In the APR+, the PARs provide the means of controlling the global hydrogen concentration in the containment while the igniter controls the local hydrogen concentration. In the sequences having a cycling relief valve leakage rate, i.e., a rate characteristic of a cycling POSRV (TRAN), both the PARs and igniters are required to maintain the hydrogen concentration in the containment below 10% by volume hydrogen. These sequences in which the PARs and igniters are operating are classified as HMS. Other sequences in which the PARs or igniters are unavailable are grouped into NOHMS."

But in the discussion of Event 5 HMS of the DET for *Mode of Early Containment Failure* the statement is made that "The igniters **and/or** PARs can prevent the potential for destructive hydrogen detonations within the containment." Please clarify if either or both igniters and PARs are required in the PRA to prevent detonations.

Response to RAI 6:

It's a typo.

In the APR+ Level 2 PRA model, BOTH igniters and PARs are assumed to be required to prevent detonations.

7. CS failure probability due to excessive debris in the dry cavity condition:

In the quantification of Event 4 CSDEBRIS: *Excessive Debris Causes CS Failure* of the DET for *Late Containment Heat Removal Status* containment spray pump failure due to core debris expelled from the cavity is discussed and P(NOFAIL) is set equal to 0.99 for all cases. This probability seems reasonable for a wet cavity, but high for dry cavity conditions. Please explain if the cavity is always assumed to be wet, or if the probabilities are used for dry conditions as well. The cavity condition (wet or dry) is ascertained for the DET for *Debris Coolability in the Cavity (CET Event: DBCOOL)* so why not for CS failure?

Response to RAI 7:

The probability of 0.01 is used for CSDEBRIS: Excessive Debris Causes CS Failure for both wet and dry cavity conditions.

The probability assignment is purely based on engineering judgment to consider potential contribution of CS pump failure due to expelled core debris to containment failure because at the time of the analysis the information on which the probability is based was not available.

In the Level 2 PRA, the wet cavity condition means the cavity is filled with water over the long term. As you mentioned in the question, debris coolability in the cavity in long term is affected by the cavity condition (wet or dry). However, it was assumed that the potential causes of CS pump failure do not depend much on cavity condition over the long term.

As a matter of fact, it's very difficult to say which cavity condition is more favorable for CS pumps failure. As described in the report, we consider two mechanisms which cause CS pumps failure. Each mechanism corresponds to each different cavity condition.

8. DCH probability in high and medium RCS pressure regimes:

The discussion on DCH failure probabilities for EVENT 8 ECFDCH: Containment *Failure due to DCH* for the DET for *Mode of Early Containment Failure* assigns a probability of 1.E- 3 to DCH which is split evenly between leak and rupture failures. This probability appears to be assigned for all RCS pressures greater that 250 psig regardless of the actual pressure in the sequence. Please explain why the effect of different pressure regimes (at least high and medium) was not considered in the quantification. Also, what is the representative high pressure accident sequence mentioned in the discussion? And what is the RCS pressure for that sequence?

Response to RAI 8:

Based on the SKN34 severe accident analysis report, the threat of containment failure due to DCH is negligible. Thus, we didn't extinguish RCS pressure regimes for DCH-induced containment failure probability estimation because the conditional probability for DCH-induced containment failure is sufficiently low and the impact on containment failure frequency is insignificant.

The representative sequence for DCH analysis is SLOCA with repressurization. The RCS pressure at the time of vessel failure is not clearly specified in the SKN34 severe accident analysis report. The report indicates that the DCH analysis for SKN34 uses NUREG/CR-6075 methodology.

9. Explanation on P(DDTOK) and justification of HMS installation:

The probability of containment failure due to a hydrogen detonation is defined in the report for Event 6 in the ECF DET as P(RUPTDET) = P(DDTOK) x P(DEF) x P(FAIL). Since P(DEF) and P(FAIL) are always 1.0, P(RUPTDET) is always determined by P(DDTOK). The values of .01 and .05 are given for P(DDTOK) for release points into the containment and the IRWST respectively when the HMS is not working. Please explain why the probability for the IRWST release is still only 0.05.

Response to RAI 9:

The probability of the DDT occurrence for the release point into IRWST is based on CE-SSAR Level 2 PRA and it is thus based on engineering judgment.

At the time of the analysis, the specific or generic information on which the probability is based was not available. The analysis-based P(DDTOK) will be determined based on the SKN34 or APR+ specific severe accident analysis report when either of the reports is available.

10. Hydrogen ignition probability after vessel breach:

In the discussion on ignition sources for EVENT 9 LH2BURN: *Hydrogen Burn After Vessel Breach* for the ECF DET the statement is made that "Even though an ignition source is assumed to exist in the containment, the probability that the ignition source actually ignites a combustible containment hydrogen mixture after vessel breach is uncertain. This parameter was nominally set at 1.0 considering uncertainty and conservatism for the condition where global hydrogen burn can be credited. Why set ignition probability to 1.0?

Response to RAI 10:

It seems that the description in the report below better matches to the probability of 0.5 (intermediate, very uncertain) in the table for analyst's degree of belief. However, as is mentioned in response to RAI 1, we want to introduce some conservatism for phenomena with high uncertainty if the impact of the conservatism on final result is not significant. Therefore, finally we selected 1.0 based on this concept.

"Even though an ignition source is assumed to exist in the containment, the probability that the ignition source actually ignites a combustible containment hydrogen mixture after vessel breach is uncertain."

At the time of the analysis, specific or generic information on which the probability is based was not available. A more realistic analysis-based probability will be determined based on the SKN34 or APR+ specific severe accident analysis report when either of the reports is available.

11. Debris coolability in the wet cavity condition:

The discussion of EVENT 5 DBCOOL: *Debris Pool Coolability* for the DET for *Debris Coolability in the Cavity* is somewhat confusing. Where is the water coming from for some of the DBCOOLW conditions? Please clarify. Also, why is the probability for cooling the debris somewhat arbitrarily split 50/50 after a long discussion on the considerations involved?

Response to RAI 11:

Debris pool coolability is an issue of severe accident phenomena over the long term. There are many research results as we mentioned in the report.

We considered two flow paths into reactor cavity which can enable successful debris cooling. First one is the flow from IRWST to reactor cavity via Holdup Volume Tank using cavity flooding system. The second one is flow from safety injection system into reactor cavity via reactor vessel after reactor vessel failure. Before reactor vessel failure, safety injection flow has not been established due to high RCS pressure or loss of power.

The probability of successful debris pool coolability is set to 0.50 by engineering judgment because 1) this phenomenon is very uncertain and 2) we want to introduce some conservatism considering uncertainty.

The analysis-based probability will be determined based on the SKN34 or APR+ specific severe accident analysis report when either of the reports is available.

12. Uncertainty analysis (methodology):

Please expand the discussion of Section 3.6 “Level II Uncertainty Analysis.” Please elaborate on the statement: “Zero/One sampling is performed for each DET branches and the simulation number is 10,000.”

Response to RAI 12:

You can see the detailed description for level 2 uncertainty analysis in the file attached. (Attachment 01)

Response to RAI-II for the Korean PSA (Round 2)

II-1. Explanation on a design objective for the APR+ nuclear power plant:

One of APR+ design objectives was given as the following sentence (in page 2).

"A cumulative frequency should be less than 1.0E-7 per reactor year for sequences resulting in greater than 0.01Sv for 24 hours at site boundary from any individual reactor."

Please give answers to the following questions.

a. Rationale for the choice (including historical background if there exists)
b. Technical implication of the criteria (0.01Sv for 24 hours)
c. Connection with commonly used L2 PSA risk measures such as CFF, LERF, LRF
d. Relation to the source term categorization

Response to RAI II-1:

a. Rationale for the choice (including historical background if there exists)

The above design objective was established for the APR+ and was basically derived from those of the APR1400, EPRI URD and CE System 80+ SSAR (CE-SSAR). The objective of the APR1400 is given as the following:

"To demonstrate that the detailed plant design with the plant located at a representative site will be capable meeting the public risk requirement; that is, that the cumulative frequency be less than 1.0E-06 per reactor year for sequences resulting in greater than 1 rem effective dose equivalent (EDE) for 24 hours at site boundary from any individual reactor."

The difference between the objectives of the APR1400 and EPRI URD (or CE-SSAR) is an amount of offsite dose which was defined as "25 rem whole body dose" in EPRI URD and CE-SSAR.

For the APR+, on the other hand, there are the two Top Tier Requirements (TTRs) related to core damage frequency from internal events, and are given as follows:

1) Core damage frequency from internal events should be less than 1.E-6/ry.
2) Core damage frequency from a single core damage accident sequence due to an internal event should be less than 1.E-7/ry.

Based on the above two TTRs, the frequency of large release to the environment in the above statement was decided to be set to 1.E-7/ry.

b. Technical implication of the criteria (0.01Sv for 24 hours)

The offsite dose of 0.01 Sv seemed to be based on the threshold value for sheltering or evacuation of US EPA PAG.

c. Connection with commonly used L2 PSA risk measures such as CFF, LERF, LRF

In Korea, there are no clear definitions for probabilistic target values of CFF, LRF and LERF in a regulatory point of view.

As mentioned in the response of RAI II-1 a), the design objective of the APR+ was determined based on core damage frequency from internal events only. From this objective, therefore, a design objective value of CFF for internal events seems to be defined as 1.E-7/yr. A design objective value of LRF for internal events can be assumed to be conservatively same as that of CFF until its definition is given. LERF can be estimated by summing the frequencies of BYPASS, NOTISO and ECF sequences according the PSAs performed in Korea.

d. Relation to the source term categorization

The above design objective was not considered in performing the APR+ Level 2 PSA, including source term categorization.

II-2. Relevance of the design objective mentioned in question II-1 :

In Section 1.1 “Objectives and Scope” there is a requirement mentioned which states: “To demonstrate that the detailed plant design with plant located at a representative site will be capable of meeting the public risk requirement; that is, the cumulative frequency be less than 1.0E-07 per reactor for sequences resulting in greater than 0.01 Sv [i.e., 1 rem] for 24 hours at site boundary from any individual reactor. What is the relevance of this requirement to the report being reviewed considering that no offsite dose calculations are performed in this report?

Response to RAI II-2:

Because the MAAP parameter file for the APR+ is under development, the evaluation of source term release to the environment is still not possible including offsite dose calculation.

The CFF from internal events, which was estimated to be 9.15E-8/ry, was less than 1.E-7/ry so that the above design objective would be satisfied.

II-3. Reference to the ASME/ANS Level 1 PSA standard:

In Section 1.2 “Methodology” it is stated that the “Level 1 portion of the analysis is equivalent to the baseline PSA described in PSA Procedures Guide (Reference 2).” Why is there no reference to the ASME/ANS Level 1 PRA Standard or to Regulatory Guide 1.200?
Please give a reason why the ASME Standard was not considered in performing the Level 1 analysis.

Response to RAI II-3:

In Korea there are no requirements for technical adequacy of PSA such as the ASME/ANS PRA Standards.

II-4. Definitions of the mission time used in Level 1 and 2 PSA:

In Section 1.3 “Analysis Ground rules” there is a statement that the “The analysis mission time for all sequences was 24 hours.” Please clarify if this time limitation applies to both the Level 1 and the Level 2 portion of the PRA. While the restriction of mission time for the Level 2 portion may be conservative in some aspects, e.g., power recovery, it could also be non-conservative in other areas. For example, on p.208 in the section on late containment failure due to containment seal failures, there is an assumption that “seal failures will not cause a failure of the containment prior to 24 hours…” Please clarify the basis of the restriction on mission time for the Level 2 portion of the PRA, and whether it is based only on the EPRI ALWR URD. Would you expect an increase in mission time to 48 or 72 hours to significantly impact the results?

Response to RAI II-4:

For systems modeling, we have mission time of 24 hours for all SSCs based on the EPRI ALWR URD.

For some containment challenges which ultimately cause containment failure e.g. late containment overpressurization (due to loss of containment spray), they are assumed to lead to containment failure with a probability of 1.0 even though containment pressure load is not sufficient to fail containment integrity within 72 hrs.

The impact of the increase in mission time on the PSA results is expected to be small.

II-5. Recovery actions credited in Level 2 PSA:

The last bullet in Section 1.3 states “Recovery actions were credited only for prevention of core damage. Recovery actions to arrest the progression of a severe accident after the onset of core damage are feasible. These include actions such as recovery of 4.16 kV power after the onset of core damage but before vessel failure.” Please clarify if feasible recovery actions in the level 2 portion of the PRA were also credited in the analysis. For example, late power recovery seems to have been credited in the DBCOOL DET (Figure 3.2-11), which would appear to contradict the first sentence of the statement quoted above.

Response to RAI II-5:

We need to revise the statement “Recovery actions were credited only for prevention of core damage.”

Off-site power recovery after core damage (before vessel failure and/or before containment failure) is the only recovery action considered in Level 2 PRA in addition to the recovery actions credited for prevention of core damage. It is incorporated into the model by adding ET headings in SBO PDS Event tree.

II-6. Sensitivity analysis on the RCP pump seal LOCA probability:

In the description of RCSFAIL DET, you explained that it was introduced to perform sensitivity analysis. Regarding to the analysis, please provide the following additional information.

a. Sensitivity analysis results when the probability equals 0.0, 0.5 and 1.0.
b. RCP seal design specification
c. Plans for the development of RCP seal model.

Response to RAI II-6:

a. Sensitivity analysis results when the probability equals 0.0, 0.5 and 1.0.

Case a: Seal Failure Probability is 0.0 (Base Case)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.07E-07	89.8
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	8.96E-09	1.0
	CF Sum	9.15E-08	10.2

Case B: Seal Failure Probability is 0.5

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.07E-07	89.9
CF	BYPASS	1.80E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.18E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	8.97E-09	1.0
	CF Sum	9.12E-08	10.1

Case C: Seal Failure Probability is 1.0

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.08E-07	89.9
CF	BYPASS	1.77E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.21E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	8.97E-09	1.0
	CF Sum	9.09E-08	10.1

b. RCP seal design specification

We have had no sufficient design information to incorporate RCP seal LOCA into Level 2 PRA model.

c. Plans for the development of RCP seal model.

As soon as sufficient design information for RCP seal is available, RCP seal model will be developed and incorporated into Level 2 PRA model.

BNL RAIs for Level 2 PSA of APR+ (Round 3)

III-1. Branches in MELTSTOP DET:

Regarding to MELTSTOP DET, please provide the following information:

a. Specific containment failure mode (Leak, rupture) for all DET sequences (1~8)
b. The reason not considering containment leak failure mode even in the sequence 6 (CHR available)

Response to RAI III-1:

a. As is shown in General CET, the MELSTOP DET classifies PDS sequences into three categories. They are CTMTFAIL (Containment Fail), MELTSTOP (Core Melt Stop) and RVRUPTURE (Reactor Vessel Rupture). Each sequence (end state) of MELTSTOP DET falls into one of the categories. Among the nine (9) sequences in MELTSTOP DET, only the sequence 07 is defined as CFBRB and it is not necessary to distinguish failure size for the containment failure mode. All the other sequences should be further analyzed in the subsequence CET Headings (corresponding DETs). Below is the table which summarizes the end state of each sequence in MELTSTOP DET.

Seq. No.	Assignment	End State
1	“RVRUPTURE”	Reactor Vessel Failure. Further Analysis will be performed in the subsequent DETs.
2	“RVRUPTURE”	Reactor Vessel Failure. Further Analysis will be performed in the subsequent DETs.
3	“MELTSTOP”	Core Melt Stop. Further Analysis will be performed in the subsequent DETs.
4	“RVRUPTURE”	Reactor Vessel Failure. Further Analysis will be performed in the subsequent DETs.
5	“MELTSTOP”	Core Melt Stop. Further Analysis will be performed in the subsequent DETs.
6	“RVRUPTURE”	Reactor Vessel Failure. Further Analysis will be performed in the subsequent DETs.
7	“CTMTFAIL”	Containment failure mode of CFBRB. It is assumed to be as a kind of rupture failure mode.
8	“RVRUPTURE”	Reactor Vessel Failure. Further Analysis will be performed in the subsequent DETs.
9	“RVRUPTURE”	Reactor Vessel Failure. Further Analysis will be performed in the subsequent DETs.

b. The end state of sequence 6 is “RVRUPTURE” in the MELTSTOP DET. “RVRUPTURE” means that the RPV breaches due to core melt before containment failure. It does not mean containment rupture failure. In the subsequent CET/DET process, the “RVRUPTURE” end states in MELTSTOP DET could be propagated to various containment failure mode like ECF(rupture), ECF(leak), LCF(rupture), LCF(leak), BMT, and NOCF.

III-2. Evaluation of ECSBS unavailability:

Regarding to the ECSBS unavailability assessment mentioned in the answer to RAI I-4 “Probability of non-recoverable failure of containment sprays”, only short description was given referring to level 1 assessment. Please provide following additional information:

a. Procedure and data used to evaluate ECSBS unavailability in Level 1 internal event PRA (including human failure event probability evaluation).
b. Factors considered that can affect human failure event probability or equipment availability for the ECSBS operation after the onset of core damage.
c. Sensitivity analysis for the parameter: CSRECSBS in MELTSTOP and CSLATE DET

Response to RAI III-2:

a. For the APR+, specific EOGs and SAMGs are not available, which are under development by KHNP. In the APR+, several calculations and documents were made to evaluate the ECSBS unavailability but these documents were not written in English, but in Korean. In the analysis, we assumed proper procedures would be prepared for ECSBS operation. Based on the calculations and design documents of ECSBS, the ECSBS unavailability was evaluated.
We provide brief results for ECSBS unavailability calculation which are presented at Figure 2-1, Table 2-1 and Table 2-2. The figure 2-1 shows the Fault Tree (Logic) for ECSBS unavailability, the Table 2-1 presents the database of each basic event for ECSBS, and the Table 2-2 presents the calculations of HFE (Human Failure Event) Probability for ECSBS operation.

b. The calculated unavailability of ECSBS is 8.00E-02 with related HFE value, 1.45E-02, in APR+ Level 1 internal event PRA. In level 1 PRA time window, core remains intact. However, in Level 2 PRA, ECSBS is considered to be operated under severe accident condition (after core damage). At the time of ECSBS operation in level 2 PRA time window, the environment for ECSBS operation becomes much worse due to core damage. It could be assumed that the probability of HFE becomes greater than 1.45E-02 (which was evaluated in Level 1 PRA) because of high stress on operators. Because the ECSBS is a dedicated system for use during severe accident, at least 24 hours after accident initiation, the probability of HFE is assumed to be approximately 2 or more times greater than 1.45E-02. And then, the branch probability of CSRECSBS in MELTSTOP and CSLATE DET is considered to be around 0.1.

c. Sensitivity analysis results when the probability equals 0.1, 0.0 and 1.0.

Case A: CSRECSBS Unavailability is 0.1 (Base Case)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.07E-07	89.8
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6

	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	8.96E-09	1.0
	CF Sum	9.15E-08	10.2

Case B: CSRECSBS Unavailability is 1.0 (the ECSBS is assumed to always fail)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	4.40E-07	48.9
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.15E-09	0.6
	CFBRB	1.36E-08	1.5
	LCF	4.10E-07	45.6
	BMT	8.47E-09	0.9
	CF Sum	4.59E-07	51.1

Case C: CSRECSBS Unavailability is 0.0 (the ECSBS is assumed to always succeed)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.48E-07	94.4
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	5.87E-09	0.7
	LCF	9.04E-09	1.0
	BMT	9.02E-09	1.0
	CF Sum	5.06E-08	5.6

Figure 2-1 Fault Tree (Logic) for ECSBS

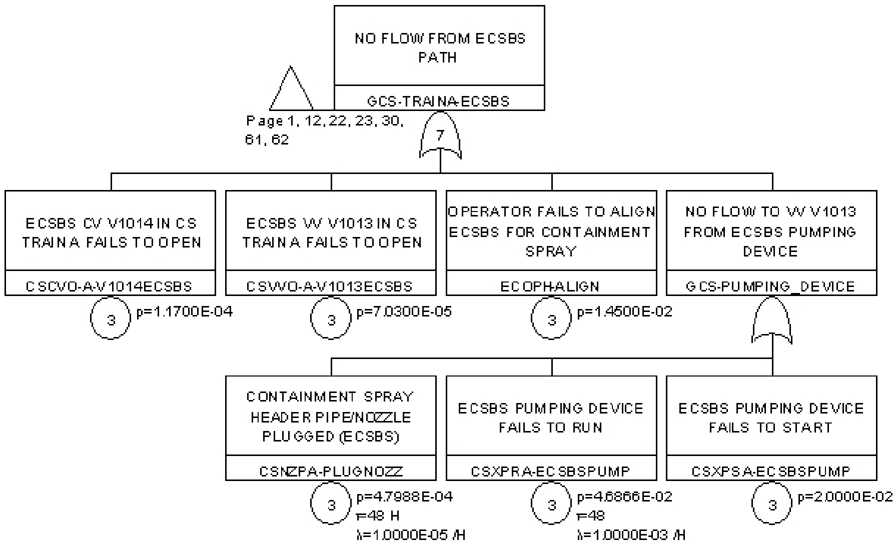


Table 2-2. Database (failure rate) for Basic Events of ECSBS

BASIC EVENT	COMPONENT (FAILURE MODE)	FAILURE RATE (generic)	ERROR FACTOR (generic)	FAILURE RATE (specific)	ERROR FACTOR (specific)
CSCVO-A-V1014ECSBS	CHECK V/V (FAIL TO OPEN)	2.00E-04 (/d) *EPRI URD	3.19 *NUCLRARR	2.00E-04 (/d)**	2.22**
CSVVO-A-V1013ECSBS	MANUAL V/V (FAIL TO OPEN)	3.88E-04 (/d) *NUCLRARR	15.00 *NUCLRARR	7.03E-05 (/d)**	5.89**
CSNZP-A-PLUG-NOZZ	Spray Nozzle (fail to operate, plug)	1.00E-05 (/h) *EPRI URD	4.84 *NUCLRARR	1.00E-05 (/h)	4.84
CSXPR-A-ECSBS-PUMP	ECSBS MDP (fail to run)	1.00E-03 (/h) *CESSAR, EPRI-URD	7.27 *NUCLRARR	1.00E-03 (/h)	7.27
CSXPS-A-ECSBS-PUMP	ECSBS MDP (fail to start)	2.00E-02 (/d) *CESSAR, EPRI-URD	10.00 *NUCLRARR	2.00E-02 (/d)	10.00

Note ** means Korean operation experience was reflected using Bayesian update process.

Table 3. Calculations of HFE (Human Failure Event) Probability for ECSBS

BASIC EVENT	HUMAN FAILURE PROBABILITY	ERROR FACTOR	TYPE
ECOPH-S-ALIGN	1.45E-02	5.0	Post-Initiator HRA

Basic Event ID :	ECOPH-ALIGN		
Basic Event Description :	Operator fails to align ECSBS for containment spray		
Data Type :	Operator Action Error		
Model(s) :	N/A		
Event Type (check one) :			
	Random-operating		Random-demand
	CCF-operating		CCF-demand
	Developed Event-operation		Developed Event-demand
	Developed Event-operating-CCF		Developed Event-demand-CCF
V	Human Error		Initiating Event
	Others(describe below)		Test/Maintenance Unavailability
CODES			
System :	N/A		
Component Type :	N/A		
Failure Mode :	N/A		
Location :	N/A		
PROBABILITY CALCULATION SUMMARY			
Mean :	1.45E-02		
Median :			
5th Percentile :			
95th Percentile :			
Error Factor :	5.0		

Basic Event ID :	ECOPH-ALIGN
CALCULATION (use additional sheets if needed):	
Description :	Operator fails to align ECSBS for containment spray
Initiating Event :	MLOCA, LLOCA
Procedures :	CS SYSTEM MANUAL(PROCEDURE)
Operator Actions	
1	Diagnose that ECSBS operation is required when the normal spraying is impossible using the CS & SC system
2	Prepare the ECSBS System (call for alter. Water source and alter. Pump)
3	Open CS-V1013
4	Start ECSBS pump
Available Time :	24hours
Action Time :	12hours
Remarks	Long available time effects operators' diagnosis and action performed properly

Table 3. Calculations of HFE (Human Failure Event) Probability for ECSBS (Cont'd)

HEP Calculations														
	BHEP	EF	Source	mean	Stress	Action Type	Place	Proced.	Time	PSF	Multi	Dep.	CHEP	
1	2.00E-05	30.0	fig 12-4	1.70E-04	MH*	Dynamic	MCR	poor	12hrs	5.00	8.48E-04		8.48E-04	
2	1.00E-02	5.0	20-6 (1)	1.61E-02	MH*	Step-by-step	MCR	poor	12hrs*	2.00	3.23E-02	HD	1.61E-02	
3	1.00E-03	3.0	20-13 (1)	1.25E-03	MH*	Step-by-step	Local	poor	12hrs*	4.00	5.00E-03	LD	4.75E-03	
4	1.00E-03	3.0	20-13 (1)	1.25E-03	MH*	Step-by-step	Local	poor	12hrs*	4.00	5.00E-03	HD	2.50E-03	
						2	1	2	1					
	Recovery	EF	Source	mean	Stress	Action Type	Place	Proced.	Time	PSF	Multi	Dep.	CHEP	Multi
1	5.00E-02	5.0	20-22 (3)	8.07E-02	MH	Dynamic	MCR	poor	12hrs	5.00	4.03E-01	LD	4.33E-01	3.67E-04
2	5.00E-02	5.0	20-22 (3)	8.07E-02	MH	Step-by-step	MCR	poor	12hrs*	2.00	1.61E-01	HD	5.81E-01	9.37E-03
3	5.00E-02	5.0	20-22 (3)	8.07E-02	MH	Step-by-step	Local	poor	12hrs*	4.00	3.23E-01	HD	6.61E-01	3.14E-03
4	5.00E-02	5.0	20-22 (3)	8.07E-02	MH	Step-by-step	Local	poor	12hrs*	4.00	3.23E-01	HD	6.61E-01	1.65E-03
						2	1	2	1				Total	1.45E-02

III-3. CS failure probabilities due to excessive debris:

In the answer to RAI I-7, it was explained that CS failure probability was based on engineering judgment and does not depend on cavity conditions. Please provide following additional information on that.

a. Items considered in the engineering judgment. To support the contention that conditions in the cavity can have minimal influence on the survivability of the CS pumps, include a discussion on the physical layout of the flow path from the cavity to the suction of the containment spray.
b. Sensitivity analysis for the probability P(NOFAIL) in CSLATE DET.

Response to RAI III-3:

a. As shown in Figure 3-1, the cavity of the APR+ is designed to minimize debris entrainment and subsequent debris dispersal into the upper compartment of the containment. The primary steam exits via a convoluted pathway above the top of the core debris chamber and into the cavity access area and out through louvered vents under the regenerative heat exchanger room. As a consequence the dominant hot gas and corium carryover pathway will be to the lower portion of the containment where the containment shell is fully protected by the secondary shield wall. It should be noted that this pathway provides limited resistance to gas flow, but is considered very deleterious for the entrained solid flow. The APR+ is also equipped with an offset core debris chamber designed to de-entrain and trap the debris ejected during a reactor vessel breach. The reactor cavity debris chamber and exit shaft have been designed such that following a failure of the reactor vessel high inertia corium debris would de-entrain and collect in the debris chamber while the lower inertia steam/hydrogen/air mixture would negotiate a right angle turn and exit the reactor cavity via a convoluted vent path. As a consequence little corium is expected to be entrained in the gas flow.

   The Figure 3-2 shows the schematic layout of the cavity, HVT, IRWST, and Containment Spray. As shown in Figure 3-2, the flow path from IRWST to Containment Spray is a circular loop as follows;

   • IRWST ⇒IRWST strainers ⇒CS pumps / Hx ⇒CS spray nozzle ⇒Trash rack ⇒ HVT ⇒IRWST

   If the cavity is flooded before reactor vessel failure, the amount of propelled debris particles would be less than that for dry reactor cavity. If the cavity is dry due to the failure of CFS, some of core debris would be finely dispersed into containment atmosphere due to DCH/HPME. As discussed above, however, debris particles propelled from the reactor cavity to the upper containment would be limited. In addition, the APR+ has IRWST sump strainers that function filtering out the debris which could be resulted in containment spray failure.

   The core debris in the cavity might be transferred into IRWST regardless of the cavity status. Considering the APR+ features described above, it is judged that the possibility of sufficient amount of fine debris to be transferred to CS pumps strainer is very unlikely (= 0.01) regardless of cavity status.

b. SI pumps failure as well as CS pumps failure due to excessive debris was considered in the sensitivity analysis. To assess the impacts of CS and SI pump failures by excessive debris on containment failure, the branch probabilities for CSDEBRIS in CSLATE DET and INJDEBRIS in DBCOOL DET are assigned as shown on the table below:

Top event	Branch	Case A (Base case)	Case B (= P(FAIL) x 10)	Case B (= P(FAIL) x 1/10)
CSDEBRIS	NOFAIL	0.99	0.9	0.999
	FAIL	0.01	0.1	0.001
INJDEBRIS	NOFAIL	0.99	0.9	0.999
	FAIL	0.01	0.1	0.001

The sensitivity analysis results are presented as follows.

Case A: CSDEBRIS/INJDEBRIS Probability is 0.01 (Base Case)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.07E-07	89.8
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	8.96E-09	1.0
	CF Sum	9.15E-08	10.2

Case B: CSDEBRIS/INJDEBRIS Probability is 0.1

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.05E-07	89.6
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	5.10E-08	5.7
	BMT	9.01E-09	1.0
	CF Sum	9.34E-08	10.4

Case C: CSDEBRIS/INJDEBRIS Probability is 0.001

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.07E-07	89.8
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.90E-08	5.4
	BMT	8.96E-09	1.0
	CF Sum	9.13E-08	10.2

Figure 3-1 Convoluted Debris Flow Path

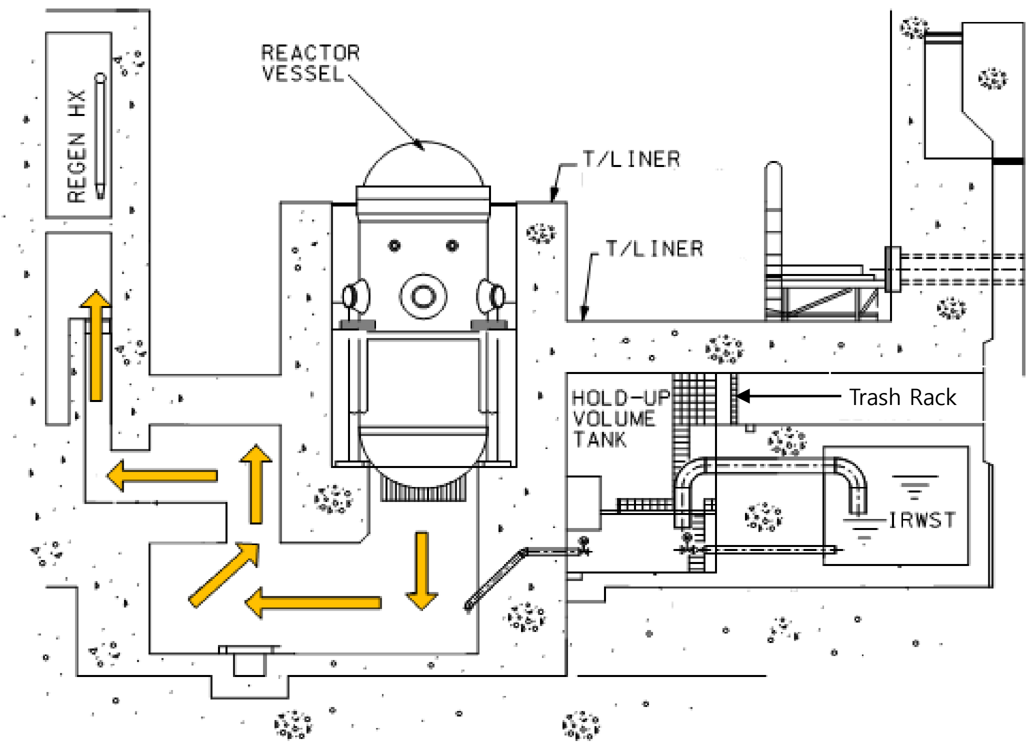
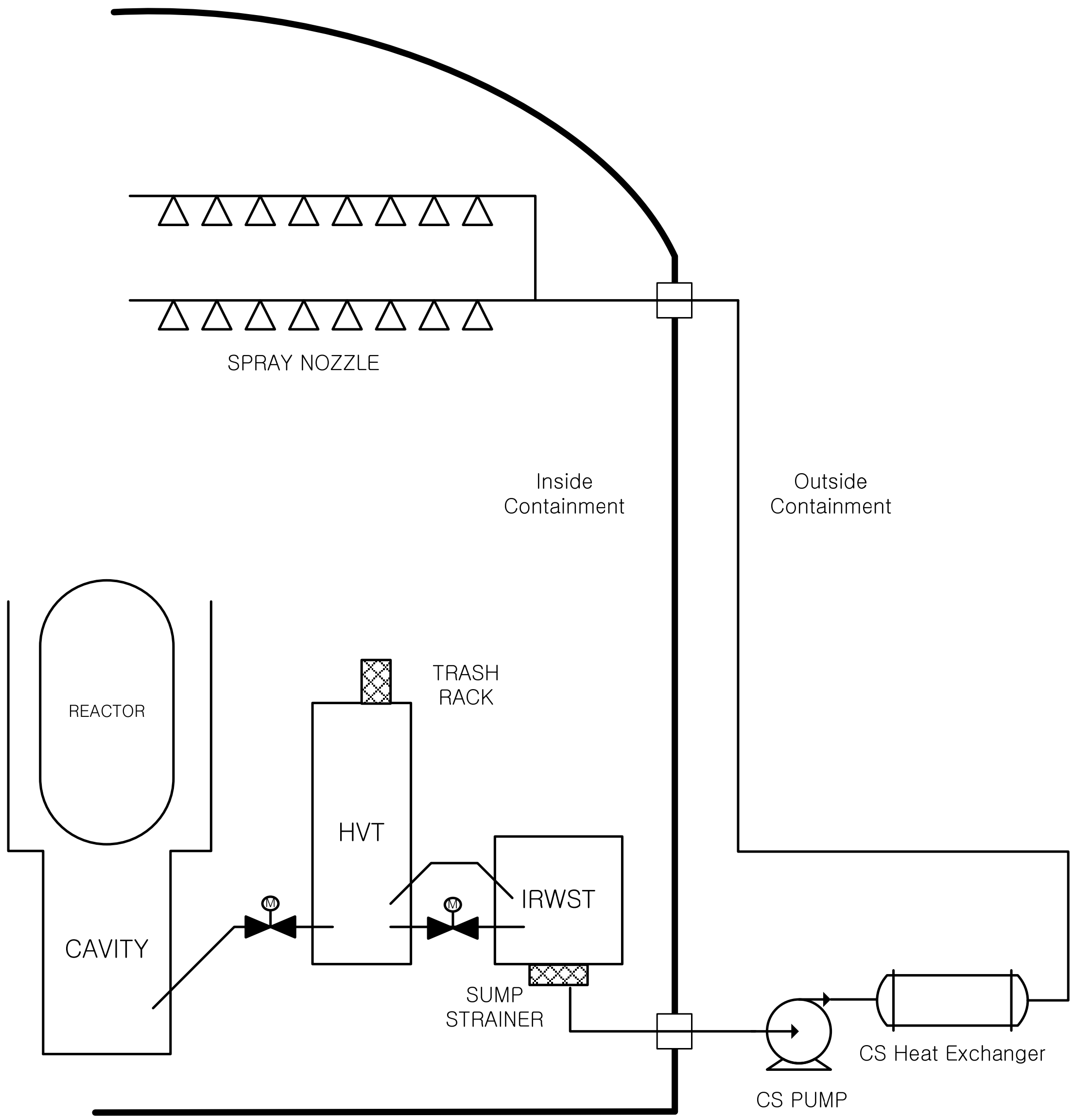


Figure 3-2 Schematic Layout of Cavity, HVT, IRWST, and CS system



III-4. Reference of containment failure probability due to hydrogen detonation:

Regarding to the answer to RAI I-9 “Explanation on P(DDTOK) and justification of HMS installation”, please provide the original paragraph you cited in CE SSAR L2 PRA report and add that document to the reference list.

Response to RAI III-4:

P(DDTOK) is considered based on System 80+ PRA. The relative paragraph in CESSAR DC is section 19.12.2.2.6.3.1.3.2.2 as follows:

19.12.2.2.6.3.1.3.2.2 DDTOK: Containment Characteristic Favor DDT

A qualitative ranking of the DDT potential for System 80+ indicates that the potential for a deflagration to transition into a detonation is highly unlikely to impossible. For the purposes of this analysis, the probability for this element is established as follows.

P(DDTOK) = 0.05 for PDSs where the release point is "IRWST".

P(DDTOK) = 0.01 for PDSs where the release point is "INC".

P(DDTOK) = 0.0 for all other PDSs.

We will add CE SSAR L2 PRA report to the reference list in the next APR+ SSAR revision.

III-5. Sensitivity analysis for DCOOL branching probability in DBCOOL DET:

In the response to RAI I-11 “Debris coolability in the wet cavity condition”, you mentioned that DCOOL branching probability was based on engineering judgment. Please perform sensitivity analysis for that parameter and report the results.

Response to RAI III-5:

Sensitivity analysis results when the probability equals 0.5, 0.0 and 1.0 are provided as follows.

Case A: DCOOL Probability P(YES) is 0.5 (Base Case)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.07E-07	89.8
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	8.96E-09	1.0
	CF Sum	9.15E-08	10.2

Case B: DCOOL Probability P(YES) is 0.0
(The core debris is assumed never to be coolable by water in cavity)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.04E-07	89.5
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	1.18E-08	1.3
	CF Sum	9.43E-08	10.5

Case C: DCOOL Probability P(YES) is 1.0
(The core debris is assumed to be always coolable by water in cavity)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.10E-07	90.1
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	6.12E-09	0.7
	CF Sum	8.86E-08	9.9

III-6. Design objectives for the APR+ nuclear power plant:

In the response to RAI II-1 and 2, you didn't provide rigorous relationship between the dose/frequency design objective and resulting level 2 PSA assessment results. For the completeness of the report, it is recommended to use such design objectives that can be derived and justified as a result of the assessment. Considering that point, if the dose/frequency criteria defined in the report cannot be clearly justified using the result assessed, please consider to use such parameters (e.g: CCFP<0.1, LRF<1.E-7/yr) as design objective which you suggested in the answer.

Response to RAI III-6:

As the design of the APR+ evolves, the source term analysis and dose calculation at the site boundary will be performed using MAAP and MACCS-2 codes. Based on the results of such evaluations, whether the APR+ design meets the design objectives could be confirmed. At the time the Technical Report was prepared, the analysis for source term release to the environment was not performed because sufficient information was unavailable. Before the sufficient information for dose/frequency estimation is available, LRF is assumed to be the containment failure frequency. Generally, LRF would be less than containment failure frequency because most of source term categories are expected not to exceed the dose limit of the design objective.

The core damage and containment failure frequencies of internal events PSA in the APR+ design were estimated to be 8.60E-07 per year and 9.15E-08 per year, respectively. These frequencies are less than the probabilistic target values of the design objectives of the APR+ and the requirements of EPR-URD and US NRC. The CCFP for the APR+ was also estimated to be 0.106, and it is approximately the same as 0.1. In particular, the design objective related to dose at the site boundary is expected to be met in the APR+ design because the estimated containment failure frequency or LRF, 9.15E-08/yr, is less than the target value, 1.E-7/yr.

III-7. Consideration on the radioactive material attenuation for the bypass sequences

The purpose of Level 2 PSA is to estimate the frequency, magnitude and other relevant characteristics of the release of radioactive material to the environment. To achieve this goal, it is important to treat potential scrubbing of radioactive material along the pathway. However, in the assessment of APR+ design, there exist no further branches for bypass sequences (SGTR, ISLOCA) in the analysis of plant damage state, containment event tree analysis and source term categorization. Please provide the plan to include more branches for those sequences considering radioactive material scrubbing effect.

Response to RAI III-7:

In the APR+ Level 2 PSA, the scrubbing effect for bypass sequences is not considered for the conservatism and the simplicity of the analysis. To evaluate the scrubbing effect for bypass sequences, it is necessary to discuss additional aspects as follows;

(1) SGTR sequences

If the point of release was submerged under deep water pool (a minimum of 10 feet of water) and the water can be maintained subcooled, scrubbing of fission products through water pools could be effective. It is expected that fission product releases below deep water pool can reduce the releases of soluble fission products by a factor of between 50 and 80.

To confirm whether the point of release is submerged or not, it is necessary to identify (1) the availability of water on the secondary side of ruptured SG, and (2) the location of tube break. If the ruptured SG is “dry” or the auxiliary feedwater supply to the ruptured SG fails, the point of release would not be submerged. If the SG is wet but the location of tube break is high (near the top of SG U-tube), the fission products would not be scrubbed because overlying water pool above the release point would not be sufficiently deep to scrub the fission products.

In APR+ Level 1 PSA, it is assumed that the ruptured SG is not used for secondary heat removal and it would be isolated by operators to prevent the leak paths of radioactive materials to the environment, including the auxiliary feed water line to the ruptured SG. The condition of the ruptured SG could also not be easily classified from present APR+ Level 1 ETs. In APR+ Level 2 PSA, therefore, it is conservatively assumed that the ruptured SG is dry at core damage and that the scrubbing effect for SGTR core damage sequences is not effective.

If the condition of ruptured SG can be easily defined from APR+ Level 1 PSA, the scrubbing effect for SGTR sequences could be applicable in APR+ Level 2 PSA.

(2) ISLOCA sequences

For ISLOCA sequences, scrubbing of fission products also requires that the break location is submerged under deep water pool and the water can be maintained subcooled. In case of ISLOCA, water in the RCS, the SITs, and/or the IRWST with SI pumps operating could flow through the break point into the auxiliary building. Submergence of the break point could be determined by the break elevation and the availability of the water sources. To evaluate the scrubbing effect reasonably, we need detailed information such as piping routing of interfacing systems, flood analysis of auxiliary building when ISLOCA occurs, analyses for effectiveness of scrubbing and so on.

If the sufficient information is available, the scrubbing effect for ISLOCA sequences could be applicable in APR+ Level 2 PSA.

III-8 Base-mat Melt-Through (BMT) probability in BMT DET:

In BMT DET, the LHTX (the likelihood of heat transfer rate to water) branch was used as the criteria to determine whether Base-mat Melt-through occurs or not. And, the probability of LHTX to be low was allocated as the term “very unlikely”(1% for the branch probability) based on the corium-water heat transfer rate estimation. However, in the case of SWN 1,2 PSA submitted for operating license, 5%~25% melt-through probabilities were used for cavity flooded sequences based on the NUREG/CR-4551 results. Please justify the values used including submission of following additional information.

a. Factors considered in the expert judgment.
b. Sensitivity analysis for the branching probabilities

Response to RAI III-8:

a. In SWN 1&2 Level 2 PSA, BMT probability is estimated by considering the amount of corium, the thickness of the corium, and the existence of cooling water in the cavity. This analysis method is consistent with NUREG/CR-4551. The BMT probabilities of 5 and 25% for cavity flooded sequences were used under the condition that the amounts of corium which is ejected out of cavity are assumed to be “medium” and “low”, respectively. The “medium” means that the amount of 20%~40% of overall corium in core is ejected out of cavity. The “low” means that the amount of less than 20% of overall corium in core is ejected out of cavity.

   However, the plant specific severe accident analysis for SKN 3&4 was performed for several accident initiators such as LLOCA, MLOCA, SLOCA, SBO, and so on. These analyses were performed using MAAP4 and MELCOR. Referring to these analysis results, the largest depth of ablation is less than 3.5 ft within 24 hrs. Therefore, it is not expected that the ablation depth would be greater than 14 ft within 72 hrs which will lead to BMT. Therefore, it is “Very Unlikely (= 0.01)” that the BMT results from the low heat transfer rate between the corium and water when cavity is flooded.

   In APR+ Level 2 PSA, the BMT probability is estimated by considering the existence of cooling water in the cavity, debris coolability (that is assigned to 0.5 due to uncertainty), and heat transfer rate between water and corium. In APR+ Level 2 PSA, most of corium materials are assumed to be collected into the cavity because of the characteristics of the cavity design. The probability for heat transfer rate between water and corium in the cavity is determined based on the plant specific severe accident analysis for reference plants (SKN 3&4) and the CESSAR DC Level 2 PRA results. (Reference: “Severe Accident Analysis Report for Shin Kori Units 3&4, KEPCO E&C, Dec. 2009 / “CESSAR-DC, System 80+ Standard Design", ABB-CE.)

   APR+ Level 2 PSA analysis about heat transfer rate between the corium and water is consistent with CESSAR DC Level 2 PRA. The related paragraph in CESSAR DC is section 19.12.2.2.7.3.1.2 as follows:

19.12.2.2.7.3.1.2 **LOWHTXFER: Corium to Water Heat Transfer Rate Is Low**

If the cavity is filled with water, the corium will transfer some of its energy to the water with the rest of the energy involved in concrete ablation. If the heat transfer rate between the corium and the water is high, the corium will be cooled and there will be limited concrete ablation. However, if the heat transfer rate is low, less than about 0.2 Mwt/m^2, the containment shell may be penetrated before the concrete ablation is terminated. There is still a significant amount of concrete (> 18 ft.) that would need to be ablated before a basemat melt-through condition would actually be reached. However, this analysis makes the very conservative assumption that penetration of the shell constitutes containment failure. This element represents the probability that the heat transfer rate from the corium to the water overburden is low enough for the corium to penetrate the containment liner within 24 hours. Thus:

P(LOWHTXFER) = 0.01 for PDSs where the cavity condition is "WET"

P(LOWHTXFER) = 1.0 for PDSs where the cavity condition is "DRY".

This mode of containment failure, ablation through the containment shell with a water overburden, is expected to lead to a highly filtered below ground fission product release. The net consequence of this failure mode on public risk should be small. (See Sections 19.11.4.2.2.4 and 19.11.4.2.2.5)

b. The sensitivity analysis results are presented as follows.

Case A: LHTX Probability is 0.01 (Base Case)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.07E-07	89.8
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	8.96E-09	1.0
	CF Sum	9.15E-08	10.2

Case B: LHTX Probability is 0.1 (= Base Case x 10)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	7.82E-07	87.0
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	3.45E-08	3.8
	CF Sum	1.17E-07	13.0

Case C: LHTX Probability is 0.001 (= Base Case x 1/10)

	CF Mode	Freq(/ry)	Contribution (%)
NOCF	NOCF	8.10E-07	90.1
CF	BYPASS	1.83E-08	2.0
	NOTISO	3.30E-09	0.4
	ECF	5.16E-09	0.6
	CFBRB	6.64E-09	0.7
	LCF	4.91E-08	5.5
	BMT	6.40E-09	0.7
	CF Sum	8.89E-08	9.9

III-9 Containment event tree after occurring of temperature induced hot-leg failure

In the general CET shown in Figure 3.2-1, no further branches were made in the “MELTSTOP” heading after temperature induced hot-leg failure occurs in the preceding “RCSFAIL” branches. Please provide the reason neglecting in-vessel core melt arrest after the hot-leg failure.

Response to RAI III-9:

In the “RCSFAIL” branches, the RCS pressure is needed to keep high before the induced failure occurs. So, injecting water into the RPV is unavailable during this period. After “RCSFAIL” or induced hot-leg or surge line failure occurs, the RCS could be depressurized and water could be injected into the RPV by the injection systems such as the Safety Injection Systems and Safety Injection Tanks. However, in this condition where the hot leg or surge line fails due to high temperature gas flow inside the RCS resulted from cladding oxidation and decay power, the core would be severely damaged and the core geometry would be deformed by the corium relocation. The core geometry would be very unfavorable to be cooled by injected water.

Therefore, it is conservatively assumed that “MELTSTOP” needs to inject water into RPV continuously just a little time after the onset of core damage during the core geometry is favorable for cooling. In other words, “MELTSTOP” in MELTSTOP DET is assumed to be probable when continuous in-vessel injection is available at the onset of core damage and the core has coolable geometry. After the core in the RPV is severely damaged, cooling the core debris is not credited.

III-10 Late containment over-pressurization in a dry cavity condition

In the bottom of the page 201 of the report, it was explained that combined effect of RCS water vaporization and core concrete attack is insufficient to cause containment failure due to containment over-pressure (Case B). Please provide the following information supporting that.

a. Type of concrete used for the bottom of the cavity
b. Assumption on the amount of non-condensable gas generated from concrete attack
c. The supporting result which compares MAAP4 predicted result with the containment fragility curve.

Response to RAI III-10:

a. The APR+ plant is under design stage and there is no plant specific information about concrete at this time. However, considering reference plants, it is expected that the type of concrete for the APR+ would be basaltic aggregate concrete type. According to the severe accident analysis report for the reference plants (SKN 3&4), the concrete type assumed is basaltic aggregate concrete which is typical for most Korean NPPs. (Reference: "Severe Accident Analysis Report for Shin Kori Units 3&4, KEPCO E&C, Dec. 2009)

b. According to the SKN 3&4 severe accident analysis report, the contribution of non-condensable gas to over-pressurization is much smaller than that of steam to over-pressurization.

   In this analysis, there are three failure mechanisms in dry cavity condition; (1) basemat melt-through, (2) failure of penetration sealants by over-temperature and (3) over-pressurization by non-condensable gas generated from MCCI. In this analysis, (1) Basemat melt-through is assessed to be the most possible failure mechanism, however, (3) over-pressurization by non-condensable gas is assessed to be almost impossible failure mechanism in dry cavity condition.

   Therefore, for APR+ Level 2 PSA, it is assumed that the amount of non-condensable gas generated in dry cavity is not sufficient to result in containment failure by over-pressurization.

c. According to the SKN34 severe accident analysis report, the analysis is performed by MAAP4 for the following three cases which have the dry cavity conditions without containment spray operation.

   (1) SBO (without battery) with dry cavity and without spray
   (2) LLOCA with dry cavity and without spray
   (3) LOFW with dry cavity and without spray

   In this analysis, the containment pressure does not reach about 90 psia (75.3 psig) for all cases above. The analysis results for three cases are presented at Figure 10-1 through Figure 10-3. Therefore, from comparison of these results with the containment fragility curve, the probabilities of containment rupture or leak by over-pressurization seem to be negligible for the dry cavity condition.

Figure 10-1 Containment pressure in SBO with dry cavity and without spray

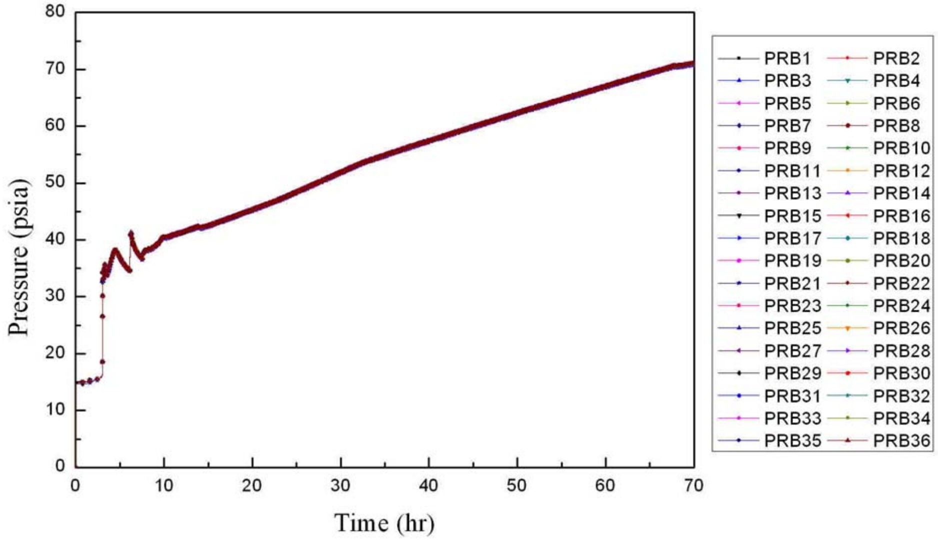


Figure 10-2 Containment pressure in LLOCA with dry cavity and without spray

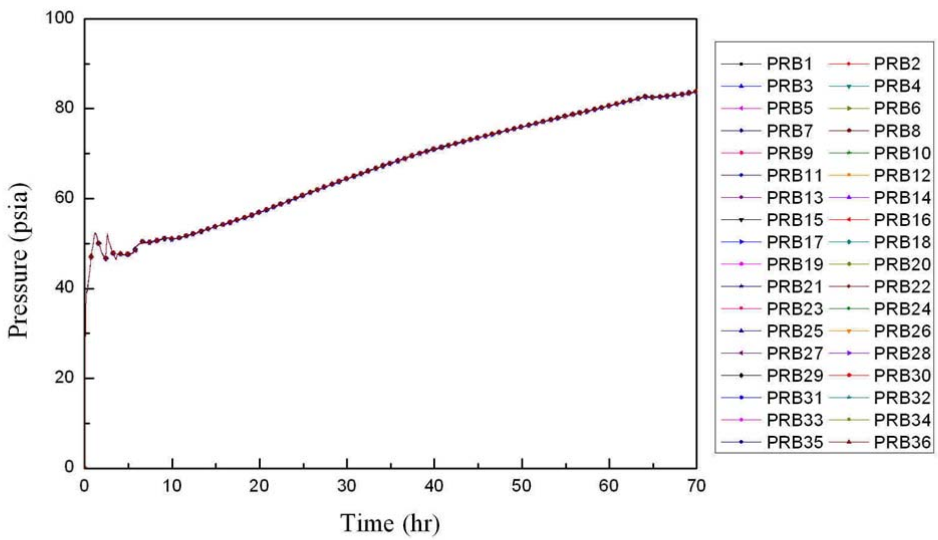
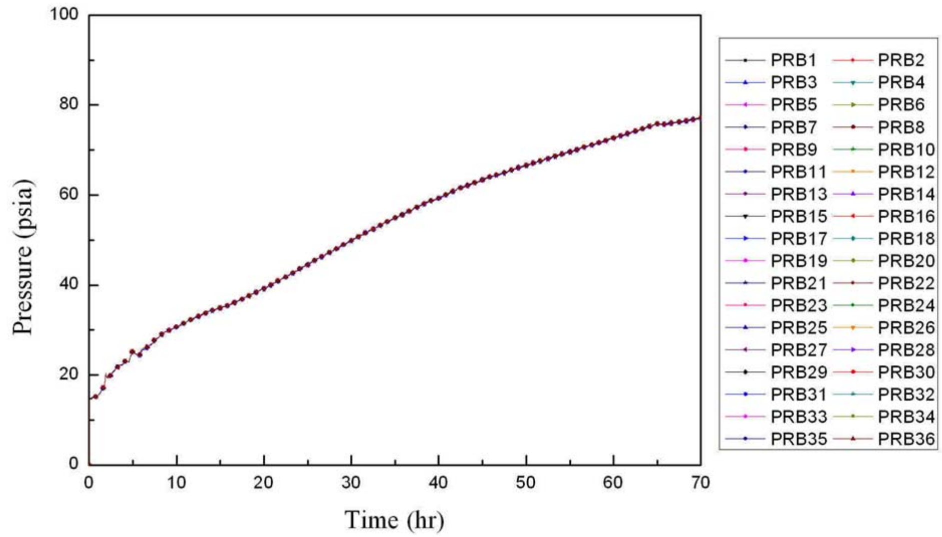


Figure 10-3 Containment pressure in LOFW with dry cavity and without spray



III-11 Consideration on secondary side heat removal status in PDS categorization

Secondary side heat removal before the onset of vessel breach or containment failure can decrease containment failure probability by reducing total heat that can be discharged into containment after the failure. However, APR+ assessment didn’t consider that effect in PDS categorization. Please provide the reason not considering the parameter.

Response to RAI III-11:

Secondary heat removal could reduce total amount of heat transfer from the core to the RPV and the inside of the containment. It could also affect severe accident progression such as delays on core damage, RPV breach, and the containment failure, including fission product scrubbing in the RCS and the SG. However, secondary side heat removal might be considered mainly to affect containment failure by gradual over-pressurization, while it hardly affects HPME/DCH, hydrogen combustion, and other containment failure modes. Reduction of total residual heat transfer to the containment could somewhat contribute to preventing containment failure by over-pressurization.

In APR+ Level 2 PSA, we considered that the containment failure by over-pressurization from the sequences with wet cavity and without containment spray is the most probable condition. It is also considered that all of these sequences, if any other containment failure modes didn’t occur, would finally result in containment failure by over-pressurization. In this condition, it is assumed that the secondary heat removal is not the matter because the effect of over-pressurization by steaming from the cavity is more dominant than that of total heat transfer reduction by secondary heat removal.

Detailed investigation into level 1 PSA sequences showed that almost all of the sequences for transient initiators failed secondary heat removal function. Therefore, assuming the secondary heat removal failure would introduce very little conservatism.

III-12 SKN 3&4 containment fragility curve

In Section 3.2.1.2.1, it was explained that the median pressure capacity of leak failure mode (188.0 psig) is lower than that of the rupture case (207.8 psig). However, in Table 3.2-1, the rupture probability (P(Rup)) for a given pressure is always greater than the leak probability (P(Leak)). Regarding to this, please provide following information.

a. Reasoning to assume rupture in advance of leak failure
b. Median value of leak and rupture failure mode if we assume “Leak before Rupture” failure mechanism according to the following procedure.
    1. Assume leak failure occurs first for a given pressure : P(Leak)
    2. For the remaining fraction, calculate P(rupture)
c. Explanation on the effect assuming that rupture is preferred mode of failure.

Response to RAI III-12:

Before responding to RAI III-12, some additional information is provided as follows.
First, the statement of “a median pressure capacity of 207.8 psig” is a typo in section 3.2.1.2.1 of APR+ PSA technical report. It should be corrected as “a median pressure capacity of 204.8 psig”
Second, it is not true that P(Rup) is always greater than P(Leak). For a given pressure between 148 psig and 186 psig, P(Rup) is smaller than P(Leak).

a. The high probability of rupture does not mean rupture occurs in advance of leak. The reason to assume rupture in advance of leak failure is that it causes conservative results in source term release to the environment. The source term with rupture failure mode is more severe than leak failure.

b. The median values of leak and rupture failure mode do not change, even if the assumption “Leak before Rupture” is made. However, P(Rup) and P(Leak) for a given pressure should be re-evaluated by using the following expressions.

$$\mathrm{Pl}(\mathrm{P}) = \int_0^{\mathrm{P}} \mathrm{f}l(\mathrm{p}')\mathrm{dp}'$$

$$\mathrm{Pr}(\mathrm{P}) = [\int_0^{\mathrm{P}} \mathrm{fr}(\mathrm{p}')\mathrm{dp}'][1 - \int_0^{\mathrm{P}} \mathrm{fl}(\mathrm{P}')\mathrm{dp}']$$

$$\mathrm{Pno}(\mathrm{P}) = 1 - [\mathrm{Pl}(\mathrm{P}) + \mathrm{Pr}(\mathrm{P})]$$

where

f_r (p): PDF (probability density function) of rupture failure at the pressure
f_l (p): PDF of leak failure at the pressure
P_r (p): cumulative distribution of rupture failure to the pressure, p
P_l (p): cumulative distribution of leak failure to the pressure, p
P_{no} (p): No containment failure to the pressure, p.

Figures 12-1 and 12-2 illustrate the fragility curves for “Rupture before Leak” and “Leak before Rupture” cases, respectively.

c. Figures 12-1 and 12-2 show that the rupture and leak failure probabilities in the pressure range up to about 145 psig are practically same for both cases, respectively. The AICC-calculated pressures of 92.4 psig and 121.4 psig (referring to

Table 3.2-2 of APR+ PSA technical report) are also in that pressure range. Therefore, the assumption of “Leak before Rupture” might make no significant differences from the original APR+ Level 2 PSA results.

Figure 12-1 Fragility curve for “Rupture before Leak “

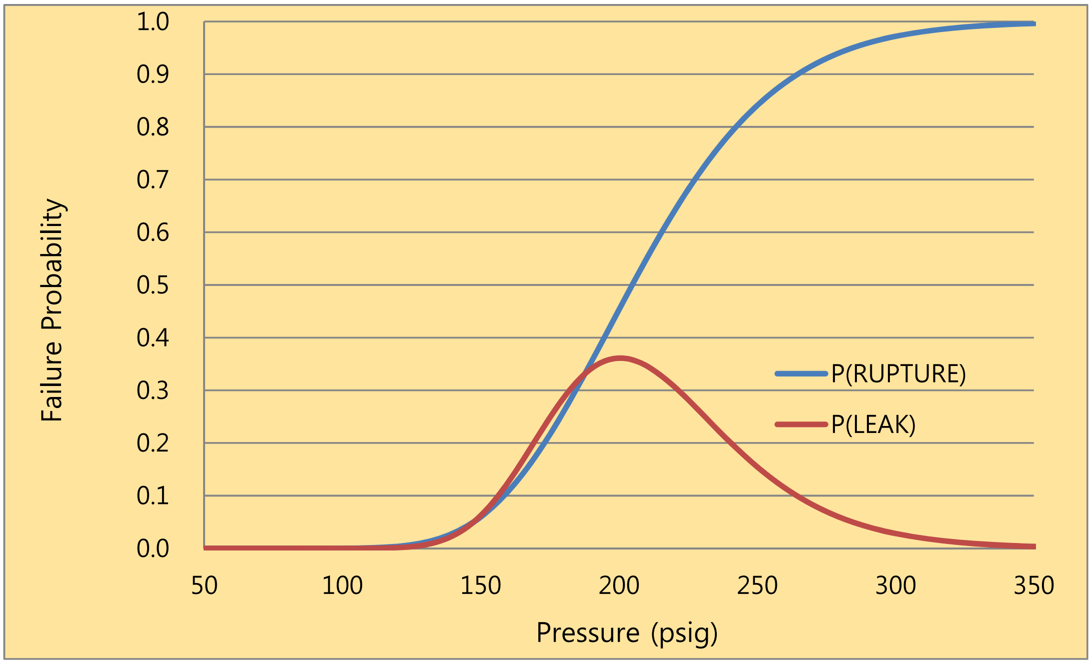
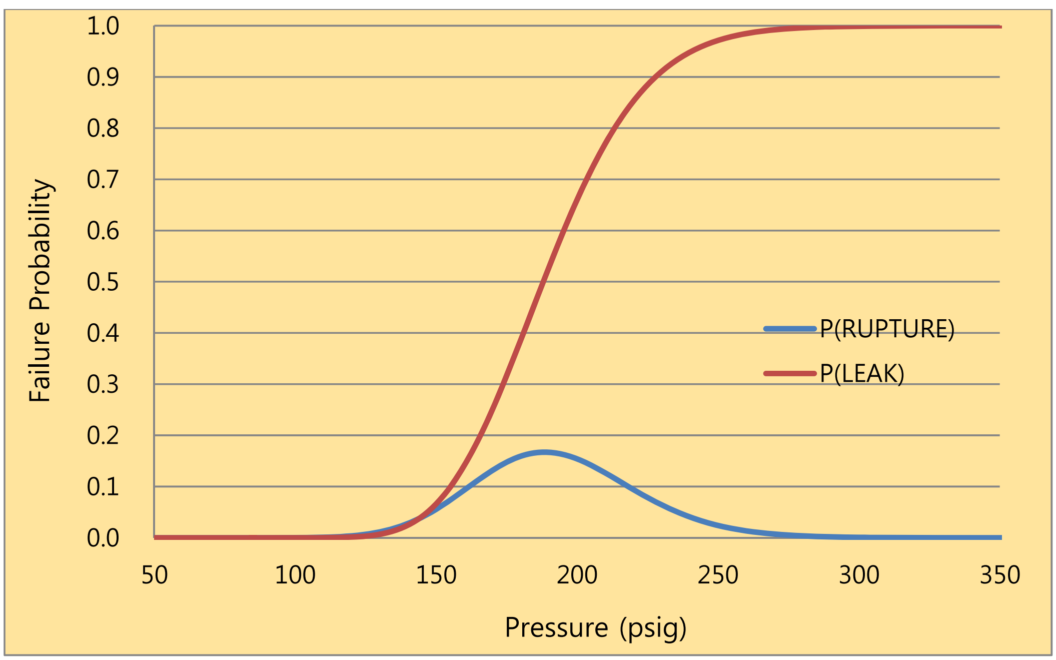


Figure 12-2 Fragility curve for “Leak before Rupture”



III-13 Assumption on the hydrogen quantity available for late hydrogen burning

In the paragraph right above Section 3.2.3.2.5 of the APR+ L2 PSA report, it is stated as

"The hydrogen burn is considered to occur when (1) HMS works properly or (2)…".

However, the last sentence of this paragraph is confusing. If we interpret your intent to be conservative correctly, we suggest that you change the sentence to something like the following: "Even though DCH or detonation occurs, it is *conservatively* assumed that the amount of hydrogen is not reduced, if the containment does not fail. This assumption is conservative because it allows more hydrogen to be available for any subsequent burns."

Response to RAI III-13:

We agree with your suggestion. Your sentence is easier to understand than original one. We will incorporate your comment.

III-14 Uncertainty analysis

Regarding to the answer of RAI I-12 “Uncertainty analysis (methodology)”, please provide more information on the following items.

a. Analysis result when we perform the simulation 1,000 times
b. Point estimated values of Table 19.1.4.6-1 and 19.1.4.6.2 don’t seem to have consistency with Table 3.2-5 and Table 3.3-1 in the APR+ L2 PSA report. Please explain the differences.

Response to RAI III-14:

a. Analysis results when performing the simulation 1,000 times are presented as follows:

CF mode	Point Estimate	Mean	Std Dev	Var	Min	5%	Median	95%	Max
NOCF	8.07E-07	8.03E-07	1.45E-07	2.11E-14	1.62E-07	4.54E-07	8.63E-07	8.64E-07	8.64E-07
ECF	5.16E-09	4.56E-09	2.82E-08	7.97E-16	0.00E+00	0.00E+00	0.00E+00	2.25E-08	5.24E-07
LCF	4.91E-08	5.13E-08	1.33E-07	1.77E-14	0.00E+00	0.00E+00	2.91E-09	4.10E-07	6.03E-07
BMT	8.96E-09	1.15E-08	5.68E-08	3.22E-15	5.75E-10	4.26E-09	7.11E-09	7.85E-09	6.15E-07
CFBRB	6.64E-09	6.70E-09	2.47E-09	6.09E-18	5.87E-09	5.87E-09	5.87E-09	1.40E-08	1.40E-08
NOTISO	3.30E-09	3.30E-09	0.00E+00	0.00E+00	3.30E-09	3.30E-09	3.30E-09	3.30E-09	3.30E-09
BYPASS	1.83E-08	1.82E-08	4.12E-09	1.70E-17	1.77E-08	1.77E-08	1.77E-08	1.77E-08	5.05E-08

b. At the time of response to the first round RAI, we presented uncertainty results using the revised APR+ level 2 PRA model which is revised for the APR+ Standard SAR. Therefore, the uncertainty analysis results for the APR+ SSAR are not consistent with those for the APR+ PSA technical report.
The uncertainty analysis results using the APR+ Level 2 PSA model for the technical report are presented as follows;

CF mode	Point Estimate	Mean	Std Dev	Var	Min	5%	Median	95%	Max
NOCF	8.07E-07	8.06E-07	1.41E-07	1.99E-14	2.61E-09	4.54E-07	8.63E-07	8.64E-07	8.64E-07
ECF	5.16E-09	5.03E-09	2.82E-08	7.96E-16	0.00E+00	0.00E+00	0.00E+00	2.25E-08	5.33E-07
LCF	4.91E-08	4.99E-08	1.31E-07	1.71E-14	0.00E+00	0.00E+00	2.91E-09	4.10E-07	6.03E-07
BMT	8.96E-09	9.74E-09	4.70E-08	2.21E-15	5.75E-10	4.26E-09	7.11E-09	7.85E-09	8.61E-07
CFBRB	6.64E-09	6.68E-09	2.44E-09	5.94E-18	5.87E-09	5.87E-09	5.87E-09	1.40E-08	1.40E-08
NOTISO	3.30E-09	3.30E-09	0.00E+00	0.00E+00	3.30E-09	3.30E-09	3.30E-09	3.30E-09	3.30E-09
BYPASS	1.83E-08	1.83E-08	4.35E-09	1.89E-17	1.77E-08	1.77E-08	1.77E-08	1.77E-08	5.05E-08