RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

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SRP Section:	**06.02.02**
Application Section:	**6.2.2**
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Question No. 06.02.02-6

SRP section 6.2.2 stipulates that, if containment pressure beyond the ambient pressure existing prior to the accident is to be credited, an evaluation of the risk associated with the possibility of inadequate containment pressure be provided. Staff guidance in Reg. Guide 1.206 for COL applicants also states that a discussion of the uncertainty involved in calculating the net positive suction head (NPSH) should be included. Currently, the DCD credits the containment atmosphere pressure as equal to the vapor pressure of the water in the in-containment refueling water storage tank (IRWST) for temperatures greater than 100° C, which results in a higher credited atmospheric pressure than was present prior to the accident. No such discussion of uncertainty or risk evaluation is provided in the DCD. This information is necessary for the staff to make a safety finding of adequate NPSH. Update the DCD to include a discussion of the uncertainty associated with the calculated NPSH, and a risk evaluation associated with inadequate containment atmosphere pressure.

Response

1. Uncertainty associated with the calculated NPSH

The uncertainty in $NPSH_r$ for the SI pumps and CS pumps is considered in determining the NPSH margin, which is the difference between $NPSH_a$ and $NPSH_r$. The required NPSH is a property of the pump and is generally determined by the pump vendor through testing. It is defined as the $NPSH_r$ to prevent a 3% loss in pump head ($NPSH_{r3\%}$) at rated flow. Following the guidance in SECY-11-0014, the following uncertainty factors associated with $NPSH_r$ are considered in determining the effective $NPSH_r$ ($NPSH_{reff}$):

$$NPSH_{reff} = (1 + \text{uncertainty})\ NPSH_{r3\%}$$

The following uncertainty factors that can affect the $NPSH_r$ developed during pump testing are considered:

a. The $NPSH_r$ varies with changes in pump speed caused by motor slip,

b. The $NPSH_r$ decreases with increasing water temperature,

c. Incorrectly designed field suction piping adversely affects the $NPSH_r$,

d. The air content of the water used in the vendor's test may be lower than that of the pumped water in the field,

e. Wear ring leakage impacts $NPSH_r$.

The $NPSH_r$ curves have not been adjusted to consider the positive impact of increasing water temperature (factor (b)); resulting in a conservative value for $NPSH_r$. A 21% total uncertainty has been applied to account for the effects of the other four uncertainty factors. This uncertainty is consistent with that used in operating plants.

Therefore:

$NPSH_{reff} = (1 + 0.21)\ NPSH_{r3\%}$

$NPSH_{margin} = NPSH_a - NPSH_{reff}$

The design basis NPSH required (effective NPSH required) for the CSPs and SIPs is specified to include the margin above the nominal $NPSH_r$ (NPSH required 3%) identified by the vendor.

A discussion of the uncertainty in $NPSH_r$ for the SI pumps and CS pumps will be added in DCD Tier 2, Subsection 6.2.2.3 (Attachment 1).

2. Risk Analysis

The risk assessment for assessing the plant risk associated with crediting containment accident pressure (CAP) in the NPSH assessment is performed using APR1400 PRA.

The methodology consists of four steps:

1) Identifying candidate initiating events/event sequences that result in loss of inventory

2) Determine sequences that can create a low pressure environment in the containment

3) Assign values to initiating event frequencies and probabilities for loss of containment isolation

4) Perform a risk assessment

The risk impact of the NPSH modeling assumption is established as follows:

$$\sum IE(CAP, Credit) * P\,\{LOI|IRWST > 212\,°F)\}$$

Where,

IE(CAP, Credit) : Initiating events requiring long term SI supply from the IRWST

P {LOI|IRWST > 212 °F)}: Probability that an unrecovered loss of containment isolation occurs prior to the long time IRWSR temperature falling to below 212 °F

The results of the risk assessment are provided in Table 1.

Table 1. Estimated Incremental Contribution to Core Damage Frequency

Initiating Event/Sequence	Event Frequency (per year)	Probability of Loss of isolation	Loss of Isolation Recovery Fails	Incremental Core Damage Frequency (per year)
Large LOCA	1.26E-06	2.98E-04	1.00E00	3.75E-10
Medium LOCA	4.85E-04	2.98E-04	2.60E-02	3.76E-09
Small LOCAs (sum of Small LOCA and fire induced small LOCA)	2.26E-03	2.98E-04	2.60E-02	1.75E-08
Internal Event induced feed and bleed (F&B) condition	9.61E-04	2.98E-04	2.60E-02	7.45E-09
Fire Induced F&B Condition	7.15E-04	2.98E-03	2.60E-02	5.54E-08
Total CDF Increase	N/A	N/A	N/A	8.45E-08

Comparing the results to the RG 1.174 incremental risk map, the total incremental core damage frequency (CDF) is considered to be a very small increment (Region I). The CDF increase may be also considered conservatively in the incremental large early release frequency (LERF) since the core damage scenarios are implicitly assumed to occur in the presence of an unisolated containment. Similarly, the incremental LERF would also be judged to be very small (Region I).

Based on the above assessment, the incremental risk impact of CAP credit based on RG 1.174 guidelines is determined to be a very small contribution to both the APR1400 CDF and LERF.

Additional details of the risk assessment for assessing the plant risk associated with crediting CAP in the NPSH assessment is provided as Attachment 3.

The summary of plant risk assessment associated with crediting CAP in the NPSH assessment will be added in DCD Tier 2, Subsection 6.2.2.3 (Attachment 1).

Impact on DCD

DCD Tier 2, Subsection 6.2.2.3 will be revised as shown in Attachment 1.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

Technical Report APR1400-E-N-NR-14001-P/NP, Section 3.6.2 will be revised as shown in Attachment 2.

mean diameter of 1,000 microns is conservatively assumed for the containment pressure and temperature analysis using GOTHIC code. The droplet of 1,000 microns has 100% effectiveness before it falls to about 12.19 m (40 ft) from the level of spray nozzles, which is shorter than the minimum spray fall height of 32.13 m (105.4 ft) (from the nozzles to the top of the PZR wall).

Containment spray elevation and plane drawings are provided in Figures 6.2.2-4 and 6.2.2-5, respectively. These drawings show spray coverage and overlap. The volume of the containment covered by the sprays is described in Table 6.5-3.

The IRWST sump strainer performance evaluation related to Generic Safety Issue (GSI) - 191 is described Reference 1 and Subsection 6.8.4.5.

NPSH

The IRWST is the suction source for the SI pumps and CSPs during short-term injection and long-term cooling modes of post-accident operation. As described in Section 6.8, the HVT performs water collection services after an accident. Spillways allow accumulated water in the HVT to spill back into the IRWST, thereby replenishing IRWST water volume during accident operations. The determination of the minimum available ~~NPSHs~~ for the SI pumps and the CSPs are based on the minimum water level in the IRWST during accident conditions. In addition, the following conservative assumptions are made:

a. ~~Fluid conditions in the IRWST are saturated; no credit is taken for an increase in containment pressure~~.

b. The contribution of the volume of water spillage from the RCS and one safety injection tank is conservatively neglected.

c. With the CSS actuated, the reactor cavity is assumed to be flooded, and the HVT is assumed to be full to just above the level at which water begins to return to the IRWST through the spillways.

d. Spray water is held up on surfaces throughout the containment. Locations of the accumulation of water inside the containment include water held up on horizontal surfaces, clogged floor drains, water held up in containment spray piping, water in

No additional containment pressure is credited above the initial containment pressure for low IRWST sump fluid temperatures (i.e., below approximately 100 °C (212 ºF). For higher IRWST sump fluid temperatures, the containment pressure is assumed to equal the saturation pressure corresponding to the sump water temperature.

the containment atmosphere, water film on vertical surfaces, puddles trapped on equipment, and the containment free volume filled with steam.

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The calculated available NPSH for the SI pumps and CSPs at the maximum expected flow, which is equal to both SI pump and CSP taking suction from the same line, is provided in Table 6.2.2-1. The calculated available NPSH exceeds the required NPSH of an SI pump and CSP from the bottom of IRWST.

A failure modes and effects analysis of the CSS is provided in Table 6.2.2-3 and demonstrates sufficient reliability of the CSS.

The integrated energy content of the containment atmosphere and IRWST water during transients is given in Part A of Table 6.2.1-38. It shows the stored energy content not only at the peak value of the containment pressure but also at the end point of transient of each phase, which includes EOB, EOR, EOPR, and 24 hours after accident initiation. Integrated energy absorbed by the structural heat sinks is also given in Table 6.2.1-38.

6.2.2.4 Tests and Inspections

Preoperational tests are conducted to verify the proper operation of the CSS. The operational tests include the calibration of instrumentation, verification of adequate pump performance, verification of the operability of all associated valves, and verification that the spray headers and spray nozzles are free of obstructions.

The CSS also undergoes preoperational hydrostatic tests conducted in accordance with ASME Section III.

Tests of individual components or the complete CSS are controlled to provide reasonable assurance that plant safety will not be jeopardized and that undesirable transients will not occur.

The preoperational test results are used to perform analyses that confirm that the as-built CSS fulfills operability requirements and provides a level of performance that meets the safety analyses.

A

Since containment accident pressure (CAP) is credited in determining available NPSH, the plant risk assessment associated with crediting CAP in the NPSH assessment was performed. The incremental risk impact of CAP credit based on RG 1.174 guidelines is estimated to be a very small contribution to both the APR1400 CDF and LERF.

The required NPSH is a property of the pump and is generally determined by the pump vendor through testing. It is defined as the $NPSH_r$ to prevent a 3% loss in pump head ($NPSH_{r3\%}$) at rated flow. Following the guidance in SECY-11-0014, the following uncertainty factors associated with $NPSH_r$ are considered in determining the effective $NPSH_r$ ($NPSH_{reff}$):

$NPSH_{reff} = (1 + \text{uncertainty})\ NPSH_{r3\%}$

The following uncertainty factors that can affect the $NPSH_r$ developed during pump testing are considered:

a. The $NPSH_r$ varies with changes in pump speed caused by motor slip,

b. The $NPSH_r$ decreases with increasing water temperature,

c. Incorrectly designed field suction piping adversely affects the $NPSH_r$,

d. The air content of the water used in the vendor's test may be lower than that of the pumped water in the plant,

e. Wear ring leakage impacts $NPSH_r$.

The $NPSH_r$ curves have not been adjusted to consider the positive impact of increasing water temperature (factor (b)); resulting in a conservative value for $NPSH_r$. A 21% total uncertainty has been applied to account for the effects of the other four uncertainty factors. This uncertainty is consistent with that used in operating plants.

Therefore:

$NPSH_{reff} = (1 + 0.21)\ NPSH_{r3\%}$

$NPSH_{margin} = NPSH_a - NPSH_{reff}$

The design basis NPSH required (effective NPSH required) for the CSPs and SIPs is specified to include the margin above the nominal $NPSH_r$ (NPSH required 3%) identified by the vendor.

suction do not change during an accident. Therefore, these system configurations result in the highest sump flow rate, which is used for sizing the IRWST sump strainers. The flow rate for the $NPSH_a$ calculation is conservatively based on the maximum pump flow rate. These calculations use Equations 2-1, 2-3, and 2-4 of Crane Technical Paper No. 410, "Flow of Fluid through Valves, Fitting, and Pipe" (Reference [3-9]) to determine the head loss due to frictional resistance in the piping and line losses due to other component. The water temperature used for head loss calculation (e.g., pipe, fitting) is 10 °C (50 °F) of the IRWST minimum temperature.

(e) Strainer head loss

The strainer head loss uses a conservative of 60.96 cm-water (2 ft-water) over the temperature of interest. The actual debris head loss is evaluated by qualified test results conducted specific to the APR1400 plant conditions. The detailed test plan is provided in Reference [3-5] and the test result is provided in Appendix C of this report. Based on the results of strainer testing, the maximum head loss for the 46.45 m^2 (500 ft^2) effective strainer area with the maximum debris load is 24.69 cm-water (0.81 ft-water) at the design flow rate and includes a clean screen component of 15.85 cm-water (0.52 ft-water). As a result of the strainer testing, a head loss of approximately 41% of the strainer design head loss ensures adequate NPSH margin for the ECCS pumps.

The strainer head loss of 60.96 cm-water (2 ft-water) represents a conservative bounding value and does not require temperature adjustment. It is conservative to use these values at higher temperatures since fluid density and viscosity decrease with increasing temperature.

(f) Required NPSH

Generally, the $NPSH_r$ (3%) is identified by the pump vendor through testing as the $NPSH_r$ to prevent a 3% loss in pump head ($NPSH_{r3\%}$) at rated flow. $NPSH_r$ is a property of the pump itself. Following the guidance in SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents" (Reference [3-10]), the following uncertainty factors associated with $NPSH_r$ are considered to determine the effective $NPSH_r$ ($NPSH_{reff}$) as follows:

$$NPSH_{reff} = (1 + \text{uncertainty})\ NPSH_{r3\%}$$

The following uncertainty factors that affect $NPSH_r$ developed during pump testing are considered:

(1) The $NPSH_r$ varies with changes in pump speed caused by motor slip.
(2) The $NPSH_r$ decreases with increasing water temperature.
(3) Incorrectly designed field suction piping adversely affects the $NPSH_r$.
(4) The air content of the water used in the vendor's test may be lower than that of the pumped water in the field.
(5) Wear ring leakage impacts $NPSH_r$.

The $NPSH_r$ curves have not been adjusted to consider the positive impact of increasing water temperature (factor (2)). This results in a conservative value for $NPSH_r$. A 21% ~~margin~~ total uncertainty has been applied to account for the effects of the other ~~three~~ four uncertainty factors. This ~~margin~~ uncertainty is consistent with that used in operating plants. The effective $NPSH_r$ of the procured pump will be confirmed through American Society of Mechanical Engineers (ASME) QME-1 qualification.

Therefore:

$$NPSH_{reff} = (1 + 0.21)\ NPSH_{r3\%}$$

Risk Assessment of Containment Accident Pressure Credit

This document provides a risk calculation for assessing the plant risk associated with crediting containment accident pressure (CAP) in the NPSH assessment.

1.0 Introduction

Limiting temperature transients indicate that post LOCA IRWST temperatures are in the range of 247-251°F. Based on calculations provided in the DCD Chapter 6 (Reference 1), these temperatures persist for a little more than a day. As the NPSH calculation credits containment pressure equal to the saturation pressure associated with the IRWST pool temperature, some fraction of the containment pressure above that corresponding to the saturation temperature of 212°F (atmospheric pressure) is credited in the NPSH calculation. This evaluation provides a bounding risk assessment of the potential impact of crediting a fraction of the CAP when IRWST temperatures are in excess of 212 °F.

2.0 Background

Adequate NPSH is essential for ensuring proper functioning of the emergency core cooling and containment spray systems. Suction for pumps in these systems is taken from the bottom of the IRWST. The IRWST is a large volume tank located inside the containment. The tank provides inventory makeup to the RCS following loss of inventory events and containment cooling for all events with steam releases to the containment. As the IRWST is located in the containment, the tank water will heat-up over time. The inventory is cooled indirectly via the containment spray heat exchanger which introduces cool water to the containment in the form of droplets which condense and cool the steam in the containment atmosphere. Over time this can lead to a condition where the containment atmosphere temperature is less than that of the IRWST water.

Analyses of LOCAs indicate that the IRWST can reach temperatures of about 250°F with a corresponding saturation pressure of about 29.9 psi. Considering this saturation temperature as the containment pressure conservatively neglects air partial pressure and otherwise implies a closed containment with a saturated steam atmosphere. Under normal design basis conditions with minimal containment leakage this model provides conservative results and there is no credible challenge to adequate NPSH. In the event that the containment cannot be maintained in an isolated condition and a large containment leak exists, it may be postulated that the air will be driven out from the containment. If this loss of overpressure occurs over a short period of time and a saturated steam environment cannot be assured or significant overcooling of the containment from additional cooling systems force a low pressure containment atmosphere, NPSH margin may be challenged. These conditions are not considered credible, but are studied within this conservatively bounding risk assessment.

3.0 Methodology

The methodology consists of four steps:

1) Identifying candidate initiating events/event sequences that result in loss of inventory
2) Determine sequences that can create a low pressure environment in the containment
3) Assign values to initiating event frequencies and probabilities for loss of containment isolation
4) Perform risk assessment

3.1 Identification of Candidate events/event sequences

Events that are considered candidates for risk impact are those events which have resulted in a loss of RCS inventory or require prolonged operation of SI pumps for long term cooling.

These events include:

a. LOCAs,
b. Events caused by single or multiple stuck open safety valves,
c. Events resulting in extended loss of steam generator heat removal that require long term operation of feed and bleed.

Available LOCA analyses are limited to design basis peak containment pressure calculations. These result in a limited but bounding information set. Both small and large LOCA scenarios result in IRWST temperatures in the range of 250°F. In risk space small LOCAs are considered as those LOCAs that span the range of 2 to 6 inch diameter breaks, with the more frequent, less challenging events (from an NPSH perspective) associated with the smaller breaks and the larger events associated with relatively low frequency breaks. LOCAs are loss of inventory events and require makeup to the RCS for an indefinite time period. In this assessment all LOCAs are conservatively assumed to require CAP credit to ensure adequate NPSH.

Events that lead to a loss of steam generator heat removal may be mitigated via the ECCS system pumps using a feed and bleed process, whereby the SI provides makeup to the RCS and a POSRV is opened to control pressure and provide steam relief. These events will also heat-up the IRWST over time, but the event may be terminated by restoration of steam generator heat removal and closure of the POSRV. Thus makeup is expected for a finite time which is scenario dependent. These sequences are also assumed to require CAP credit to ensure adequate NPSH.

Note that steam line breaks have not been included in the above list. Steam line breaks inside the containment require spray actuation to control containment pressure and SI injection to makeup inventory shrinkage and provide adequate boron to maintain the core in a shutdown condition. These requirements occur early in the transient when the SI injection and containment spray operation is supported by a cold IRWST. While the IRWST does heat-up above 212°F over time, these events will not progress to a core damage condition as without additional failures. Consequently, these sequences are not further considered in this risk assessment.

3.2 Determine sequences that can create a Low Pressure Environment in the containment

As discussed above, events that contribute to an increase in core damage probability are associated with a loss of core inventory sequence coupled with a loss of containment isolation or excessive containment cooldown. A large loss of containment isolation affecting the containment atmosphere space for an extended time will cause a loss of air. Coupled with cooling of the containment atmosphere, the NPSH conditions assumed in the design NPSH could be violated. If these conditions occur prior to the IRWST cooling to 212°F, it is conservatively assumed that these conditions will result in failure of SI. Recovery actions are assumed ineffective and core damage results.

Containment isolation failure can occur as a result of failure of containment systems to isolate following an event or be a result of pre-existing containment leaks (e.g., containment flaws). For simplicity all loss of containment isolation events associated with failure of systems to isolate are considered sufficiently large to create a loss of NPSH. Pre-existing containment flaws are rare and small. Historically the frequency of pre-existing containment leaks has been evaluated in Reference 2. Considering the duration of the IRWST temperature above 212°F will last on the order of 24 hours[1], and assuming a maximum La of 0.1 volume percent per day, the containment atmospheric leakage will vary from between 3.5 to 10 volume percent. The net impact on the containment atmosphere while not negligible, is small (less than 1.5 psi on air partial pressure, corresponding to an approximate 10% of the initial air volume over the first day) and will not impact conclusions with regard to NPSH adequacy. Therefore, this containment failure mode is not considered to be a credible contribution to SI pump failure and consequent core damage.

The second scenario that could potentially cause the containment pressure to fall below the saturation pressure corresponding to the IRWST temperature involves the addition of significant excess cooling capacity to the containment. This might be caused by a large HVAC system being actuated in addition to the two containment sprays, or possibly the spray temperatures are much lower than the minimum assumed in the containment analysis. These scenarios are not considered credible for the APR1400 design and are not further pursued.

3.3 Assignment of Initiating Event Frequencies and Loss of Isolation Probabilities

3.3.1 Loss of Coolant Accidents (LOCAs)

Initiating event frequencies (IEFs) have been established based on a combination of APR1400 PRA data and judgment. Initiating event data was directly available for LOCA and stuck open POSRV events. Data from LOCA events is summarized in Table 3-1.

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[1] The interval from the event initiation to the time the IRWST temperature reaches 212 °F following the initial peak is estimated to be 100,000 seconds (27.7 hrs) (DCD Figures 6.2.1-4 through 6.2.1-6). Beyond that time loss of isolation will not cause flashing of sump water.

Table 3-1 Summary of LOCA Initiating Event Frequency

Event Description	Frequency	Comments
Large LOCA	1.26E-06 /year	DCD Table 19.1-6
Medium LOCA	4.85E-04 /year	DCD Table 19.1-6
Small LOCA (including stuck open POSRV)	1.99E-03 /year	DCD Table 19.1-6
Fire Induced SLOCA	2.72E-04 /year	Reference 3 (Table V.1)
Total LOCA Scenarios	2.75E-03 /year	

3.3.2 Loss of Steam Generator Heat Removal Scenarios

Feed and bleed scenarios represent an intermediate PRA success state which is not directly tracked. These states arise from an event that causes a total loss of steam generator heat removal. In the APR1400 PRA, such events may arise from a variety of initiating events including General Transient, TLOCCW, LODC, LOFW, LOOP or fire induced events coupled with downstream failures of combinations of EDGs, and turbine driven/motor driven pump failures with or without intervening operator errors. Fire induced events may have some overlapping common cause failures due to the fire. In the absence of feed and bleed, these events would proceed to core damage. Feed and bleed scenarios were approximated by using available initiating event information with high level approximate estimates of subsequent failure events.

In internal event, there are several initiating events leading to feed and bleed scenarios. The frequency of initiating events and failure rate of secondary heat removal are summarized in table 3-2. The initiating events in table 3-2 include feed and bleed sequence. Some initiating events (LLOCA, MLOCA, SLOCA and PR-SL) are excepted because they are already considered as the events that increase IRWST water temperature. In this result, the sum of initiating event frequency is 8.28E-01. These events are primarily mitigated by decay heat removal through the steam generator. For purpose of this assessment it is assumed that the failure rate of secondary side decay heat removal given event is 1.16E-03/year conservatively using maximum value. Thus the IEF for these feed and bleed scenarios would be approximately 9.61E-04/year. Thus the IEF for these feed and bleed scenarios would be 9.61E-04/year for internal analysis.

Table 3-2 Summary of Initiating Event Frequency (Internal Event)

Initiating Event		SHR	**Comments**
IE	Frequency	Probability	
Steam Generator Tube Rupture	1.97E-03	6.73E-04	DCD Table 19.1-6
Partial Loss of ESW	1.63E-03	6.66E-05	
Partial Loss of CCW	2.10E-03	6.66E-05	
Large Secondary Side Break – Upstream	3.49E-04	6.72E-04	
Large Secondary Side Break – Downstream	7.32E-03	2.26E-06	
Loss of Offsite Power	2.68E-02	1.16E-03	
Loss of Instrumentation Air	7.81E-03	4.36E-06	
Loss of Feedwater	6.55E-02	1.19E-05	
Loss of 1E 125VDC vital bus B	7.00E-04	1.62E-04	
Loss of 1E 125VDC vital bus A	7.00E-04	1.62E-04	
Loss of Condenser Vacuum	5.57E-02	1.19E-05	
General Transients	6.56E-01	4.59E-06	
Feedwater Line Break	1.74E-03	6.73E-04	
-	8.28E-01 (Sum)	1.16E-03 (Maximum)	

Initiating events leading to feed and bleed scenarios were assumed all of fire induced internal events (Reference 3, Table V.1). Sum of fire induced internal event frequency is 1.43E-01. In this result, the most impact initiating event is fire induced GTRN, 1.19E-01. These events are primarily mitigated by decay heat removal through the steam generator. However, fire induced scenarios may take out diverse systems needed to support containment heat removal. For purpose of this assessment it is assumed that the failure of secondary side decay heat removal given all of fire induced internal event is less than 5.00E-03/year. Thus the IEF for these feed and bleed scenarios is 7.15E-04 /year.

3.3.3 Assignment of Loss of Isolation

Containment isolation may be lost via a failure of containment isolation valves to close. The APR1400 PRA considers loss of containment isolation probability to be 2.98E-04 (based on a general transient condition, the result is calculated using fault tree). For fire induced events this value is increased by an order of magnitude to account for potential increase in loss of isolation during these scenarios.[2] The majority of the loss of isolation event is relatively small (on the order of 2” diameter) and would not result in a significant loss of containment environment. Regardless, it is assumed that all loss of containment isolation conditions is sufficient to challenge the availability of adequate NPSH.

Based on the discussions in sections 3.3.1 and 3.3.2 the frequency of initiating events that credit containment overpressure and would be expected to be successful(assuming containment isolation in not lost) is below 4.50E-03/year. The majority of those events will be driven by LOCA initiating events and the other portion will be due to intentional inventory releases to establish a stable feed and bleed condition. Of those events a small fraction will occur in the absence of an isolated containment. This condition will allow loss of air inventory which could leave the containment vulnerable to containment pressures which could fall below the saturation pressure of the IRWST when the IRWST temperature is in excess of 212°F.

It is assumed that those events that initiate as a LOCA will drive air from the containment early in the process. For feed and bleed scenarios inventory loss is slower and a potential for restoration of containment isolation is possible by manual actions. Since the loss of air mass will take place over many hours, the likelihood of failing to restore containment isolation prior to significant loss of air mass for these events is taken from the PRA as 2.60E-02 for all scenarios except the large LOCA. No credit for containment isolation recovery is considered for the large LOCA.

4.0 Risk Quantification

Given the above assumptions, the risk impact of the NPSH modeling assumption may be established as follows:

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[2] A more realistic assessment of this parameter should be performed. This estimate is expected to be an overestimate of the impact.

$$\sum IE(CAP, Credit) * P\,\{LOI|IRWST > 212\,°F)\}$$

$$IE(CAP, Credit) = Initiating\ events\ requiring\ long\ term\ SI\ from\ the\ IRWST$$

$$P\,\{LOI|IRWST > 212\,°F)\} = Probabiliry\ that\ an\ unrecovered\ Loss\ of\ Isolation\ occurred\ prior\ to\ the\ long\ time\ IRWSR\ temperature\ falling\ to\ below\ 212\,°F$$

Calculations are provided in Table 4-1.

Table 4-1 Estimated Incremental Contribution to Core Damage Frequency

Initiating Event/Sequence	Event Frequency (per year)	Probability of Loss of isolation	Loss of Isolation Recovery Fails	Incremental Core Damage Frequency (per year)
Large LOCA	1.26E-06	2.98E-04	1.00E00	3.75E-10
Medium LOCA	4.85E-04	2.98E-04	2.60E-02	3.76E-09
Small LOCAs (sum of Small LOCA and Fire induced small LOCA)	2.26E-03	2.98E-04	2.60E-02	1.75E-08
Internal Event induced F&B condition	9.61E-04	2.98E-04	2.60E-02	7.45E-09
Fire Induced F&B Condition	7.15E-04	2.98E-03 (discussion in Section 3)	2.60E-02	5.54E-08
Total CDF Increase	N/A	N/A	N/A	8.45E-08

Based on the above assumptions and comparing the results to the RG 1.174 incremental risk map, the total incremental core damage frequency (CDF) may be considered to be very small increment (Region I). The CDF increase may conservatively be also considered the incremental large early release frequency (LERF) as the core damage scenarios are implicitly assumed to occur in the presence of an unisolated containment. Similarly, the incremental LERF would also be judged to be very small (Region I).

5.0 Discussion of Conservatisms

The above analysis provides a bounding analysis of the risk impact of CAP credit taken for APR1400. These analyses tacitly assume that all loss of isolation events can potentially result in sufficiently rapid depressurization to create significant flashing in a "hot" (> 212 °F) IRWST water pool to cause the SI pumps to fail and that failure is unrecoverable. Neither of these assumptions are likely outcomes of the loss of isolation. Most containment penetrations are very small, such that the loss of isolation will more likely result in a gradual evaporation and cooling of the pool. Furthermore, only a small fraction of these penetrations isolate the containment atmosphere. Secondly, tests on the SI pumps suggest that these pumps are relatively rugged and can operate successfully with minimal (near zero) (Reference 4).

As noted above, additional conservatisms in the analyses include assuming all small LOCA events will be subject to the same NPSH challenge, regardless of break size. Additional conservatisms are also introduced in assuming a low recovery probability for auxiliary feedwater which would reduce the contribution of the feed and bleed events to the CAP core damage estimate.

6.0 Conclusions

Based on the above assessment the incremental risk impact of CAP credit based on RG 1.174(Reference 5) guidelines is estimated to be a very small contribution to both the APR1400 CDF and LERF.

7.0 References

1 Advanced Power Reactor (APR1400) Design Certification Document, submitted via Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd Application for Design Certification of the APR1400 Standard Design, December 23, 2014.

2 EPRI-TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" Electric Power Research Institute, Revision 2-A, December, 2008 (TR-1018243).

3 APR1400-K-P-NR-013404-P, "Internal Fire PRA Volume IV", Revision 0, July, 2013.

4 "Prediction of Life of PWROG Safety Related Pumps under Cavitating Conditions Final Report (PO 4500649509)", Cooper, D., Westinghouse, September, 2014.

5 Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” U.S. Nuclear Regulatory Commission, May, 2011, ADAMS Accession No. ML100910006.