UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 14, 2016

MEMORANDUM TO: ACRS Members

FROM: Maitri Banerjee, Senior Staff Engineer **/RA/**
Technical Support Branch
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS APR1400 SUBCOMMITTEE ON SEPTEMBER 21-22, 2016, ROCKVILLE, MARYLAND

The minutes for the subject meeting were certified on November 9, 2016. Along with the transcripts and presentation materials, this is the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc with Attachment: A. Veil
M. Banks

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: Maitri Banerjee, Senior Staff Engineer
Technical Support Branch
Advisory Committee on Reactor Safeguards

FROM: Ronald Ballinger, Chairman
APR1400 Subcommittee
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFIED MINUTES OF THE ACRS APR1400 SUBCOMMITTEE MEETING ON SEPTEMBER 21-22, 2016

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on September 21-22, 2016, are an accurate record of the proceedings for that meeting.

/RA/ **November 9, 2016**

Ronald Ballinger, Chairman Dated
APR1400 Subcommittee

Certified on: November 9, 2016
Certified by; Ronald Ballinger

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE APR1400 SUBCOMMITTEE MEETING ON
SEPTEMBER 21-22, 2016, ROCKVILLE, MD

The ACRS APR1400 Subcommittee held a meeting on September 21-22, 2016 in T2B1, 11545 Rockville Pike, Rockville, Maryland. The meeting convened at 8:30 a.m. on September 21 and adjourned at 9:31 a.m. on September 22. The meeting was open to the public.

No written comments or requests for time to make oral statements were received from members of the public related to this meeting.

ATTENDEES

ACRS Members/Staff:

R. Ballinger, Chairman
P. Riccardella, Member
J. Rempe, Member
D. Skillman, Member
Matthew Sunseri, Member
Maitri Banerjee, ACRS Staff*
J. Stetkar, Member
C. Brown, Member
Jose A. March-Leuba, Member
D. Powers, Member
C. Brown, ACRS Staff (DFO)

KHNP, NRC Staff, Consultants & Other Attendees:

J. CIOCCO, NRO	WILLIAM WARD, NRO
BOB CALDWELL, NRO	JOHN BUDZYNSKI, NRO
LUISSETTE CANDELARIO, NRO	STEPHANIE DEVLIN-GILL, NRO
STEVEN DOWNEY, NRO	DAVID HEESTEL, NRO
MICHAEL LEE, NRO	JOHN HONCHARICK, NRO
CHANG-YANG LI, NRO	GREG MAKAR, NRO
MICHAEL D. MAZAIKA, NRO	ROBERT ROCHE-RIVERA, NRO
RICARDO RODRIGUEZ, NRO	TARUN ROY, NRO
JAMES STECKEL, NRO	ANGELO STUBBS, NRO
SESHAGIRI TAMMARA, NRO	VAUGHN THOMAS, NRO
JESSICA UMANA, NRO	DAN WIDREVITZ, NRO
JASON WHITE, NRO	
DAEGUN (TONY) AHN, KHNP	JIYONG(ANDY) OH, KHNP
JUNGHO KIM, KHNP	WOOCHONG CHON, KEPCO NF
SEOK JEONG YUNE, KEPCO E&C	JAMES ROSS, KHNP /AECOM
SEOKHWAN HUR, KEPCO E&C	TAE HAN KIM, KEPCO E&C
JUNG BOM JANG, KEPCO E&C	GYUN DO JEONG, KHNP
HYGOK JEONG, KEPCO E&C	JONGSOO KIM, KHNP

YONGSUN LEE, KEPCO E&C	KWANGIL LIM, KEPCO, E&C
YUNHO KIM, KHNP	KWANG-HOON KOH, KEPCO E&C
SUNGUK KWON, KHNP	DONG-SU LEE, KHNP
DUK BIN SONG, KEPCO E&C	JEONG KWAN SUH, KHNP
MINSEOK KIM, KHNP	TAE SUN RO, KEPCO- E&C
SEOGNAM CHOI, KHNP	ROB SISK, WESTINGHOUSE

*participating via telephone

SUMMARY

The purpose of the meeting was for the ACRS members to receive a briefing on Korea Hydro and Nuclear Power Company (KHNP) design certification application (DCA) and NRC staff review specific to Tier 2 Chapters 2 and 5 and related information. The meeting transcripts are attached and contain a description of each matter discussed during the meeting. The presentation slides and handouts used during the meeting are attached to these transcripts.

The following list describes significant issues discussed during the meeting with the corresponding pages in the transcript referenced. Unless specifically noted the chapter and section references belong to the DCD Tier 2 submittal or staff safety evaluation report (SER).

DAY 1: September 21, 2016

SIGNIFICANT ISSUES	
Issue	Reference Pages in Transcript
Chairman Ballinger started the meeting by introducing the ACRS members present. He noted the purpose of the meeting was to review KHNP DCA and staff review related to APR1400 Chapters 2 and 5.	6-8
After a short introduction by Mr. William Ward of NRO, Mr. Jeff Ciocco, the lead project manager for the NRC APR1400 DCA review team, started the staff presentation. Mr. Ciocco noted the staff was about a year and a half into the NRC's 42 month review schedule.	8-10
Mr. Yunho Kim, the technical project manager for APR1400, started the KHNP presentation by providing an overview of APR1400 DC project including the history of nuclear power in Korea, development of APR1400, project organization, design features and general arrangement for APR1400, DCA review status, and major differences between APR1400 and System 80+ design. He noted KEPCO and KHNP are the co-applicants of the project. Upon member Skillman's question he noted some design features of the APR Plus Design (1500 MWe) and the basis for the 16 hour station blackout (SBO) coping time for APR1400. He noted	10-26 Slides 1-16

the tests done for GSI-191.	
Mr. Kwang Hoon Koh, KEPCO E&C, started Chapter 2, Site Characteristics, presentation. Slides 30-33 contain a summary table.	28-29 Slides 1-7
Chairman Ballinger noted that NRO staff will not present Section 2.3, although KHNP will.	29-30
Mr. Dong-Su Lee, KHNP, started with outdoor air temperatures. Member Stetkar questioned the 81 degrees Fahrenheit value for the zero percent exceedance non-coincident wet bulb temperature for the essential service water system, and that it was very low to cover all US sites. He wanted to know what elements of the design limit that temperature. Mr. Rob Sisk, Westinghouse noted that they would answer at a later meeting.	30-34 55-57 Slides 8-9
Mr. Lee then addressed atmospheric dispersion factors. Member Skillman asked why Prairie Island was selected as the most limiting meteorological condition among six U.S. nuclear power plant sites for the purpose of calculating onsite design χ/Q values. Mr. Lee noted that resulting χ/Q values were increased by 50 percent to bound U.S. site meteorological data. Member Rempe asked why in their draft SER staff noted that often KHNP in Chapter 2 references the site boundary when they mean the exclusion area boundary (EAB). This was taken for follow-up.	34-38 Slides 10-12
Mr. Koh presented Section 2.4, Hydraulic Engineering. Member Stetkar asked which buildings need to be protected for ground water given the values of maximum design flood and ground water level. KHNP took this question for later follow-up.	38-40 Slides 13
Member Stetkar noted Section 2.4.4.2 (Unsteady Flow Analysis of Potential Dam Failures) references "artificially large floods," and wanted to know the meaning. He noted also, in Section 2.4.11 (Low Water Considerations) low flow in rivers and streams; low water resulting from surges, seiches, or tsunamis, were mentioned for a COL applicant to evaluate, but failure of downstream dams or other impoundments was not mentioned regarding impact on the essential cooling water supply, or ultimate heat sink, if the plant is constructed that way. He noted also that he could not find any discussion of groundwater chemistry, and if any limits need to be provided for protection of below-grade structures. KHNP took this question for later follow-up.	40-43
Mr. Koh presented DCD Section 2.5 that describes geology, seismology, and geotechnical engineering. He discussed how various response spectra are developed/used. KHNP to fix DCD Tier 2 Figures 2.0-1 and 2.0-2 (Tier 1 Figure 2.1-1 and 2.1-2) for certified seismic design response spectra by labeling the damping values.	43-46 Slides 14-18

Mr. Koh discussed the qualification of soil and subsurface material, including backfill. Member Stetkar wanted to know which buildings were in the scope of the certified design. Two responses were provided that required additional clarification. Mr. Sisk stated that this would be covered in more detail during the Chapter 3, Sections 3.7 and 3.8 presentation. Member Stetkar pointed out that COL information items in Chapter 2 may need to be clarified also. Later in the meeting Andy Oh, KHNP, clarified the ESW and CCW systems are in scope for the APR 1400 (certified) design. However, the buildings' seismic analysis is out of scope. Member Stetkar reiterated his comment that information in Chapters 2 and 3 needs to be clear so that a COL applicant understands the scope of the certified design SSCs and criteria for subsurface soil, groundwater, etc. they must meet (i.e., the scope may not be limited to nuclear island only).	46-59 Slides 19-22 59-60 74-76
Member Stetkar noted in KHNP Slide 20, the maximum allowable differential settlement between buildings is 3-inches between the nuclear island base mat and the emergency diesel generator building, and the diesel fuel oil tank building. He wondered how this number is justified given there may be piping, ventilation lines and cables passing through. A discussion followed. It was pointed out that where piping is crossing in between buildings, the allowed settlement value was much less.	50-53
Chairman Ballinger asked for a clarification regarding a statement in DCD Section 2.1.1 that states a highway, railroad, or waterway may traverse the exclusion area but not close enough to the facility to interfere with normal operations. This was taken for future follow-up	53-54 76-77
With a presentation on key RAIs and a summary, Mr. Lee concluded KHNP presentation on Chapter 2.	54-58
Mr. Tammara presented staff evaluation for Sections 2.1 and 2.2. Member Stetkar asked for a clarification regarding the staff acceptance of quantitative screening criterion to screen out design-basis events (DBE) caused by nearby industrial, transportation, military facilities and other human-related hazards, when RG 1.200 provides for adjustment to the criterion for designs with substantially lower risk profiles. It appears that staff did not have expertise available to answer the question.	60-72 Slides 8-9
Member Stetkar noted that staff SER Section 2.2.3.3 includes missiles more energetic than the tornado missile spectra, and pressure effects in excess of the design-basis tornado in types of hazards to be considered. He noted in some areas of US, particularly the SE, the hurricanes could provide conditions more stringent than tornados. He asked why hurricanes were not mentioned. Mr. Tammara referred to Section 2.3 reviewer. It appears that staff did not have expertise available to answer the question.	66-71

Mr. Mike Lee presented NRO review of Section 2.4, Hydrology. Member Stetkar repeated his question regarding which safety-related structures are within the scope of the DCD flood and ground water levels.	72-79 Slides 10-13
NRO Staff presentation on their review of Section 2.5 addressing the scope and findings. Mr. Rodriguez noted there was a recent RAI response that may have some impact into Section 2.5. Member Stetkar repeated his concern that the staff need to have COL information items in Chapters 2 and 3 SER to clearly state the scope of the DCD regarding structures like buildings and tunnels.	79-84 Slides 14-18
KHNP presentation of Chapter 5 – Mr. Minseok Kim, KHNP, discussed the reactor coolant system (RCS) functions, principal parameters, Code compliance, Overpressure Protection, Reactor Coolant Pressure Boundary (RCPB) Materials, ISI and IST, RCPB Leakage Detection, Reactor Vessel Materials, P-T limits, Pressurized Thermal Shock, Reactor Vessel Integrity, and RC Components (pumps, steam generator, pressurizer), Shutdown Cooling System, RC High Point Vents, and component supports.	85-91 Slides 1-7
Member Skillman inquired about the transients that are included in 60 year design life consideration of RCPB. He asked for information related to RCP reverse rotation conditions and on reverse flow delta T versus time that are included in the (fatigue) analysis in compliance with the Code. KHNP states that such transient was to be included in Section 3.9.1.	87-91
Member Skillman asked about the sizing basis of the reactor vessel head vent. When KHNP noted that natural circulation cooling analysis provides for the basis, Member Skillman asked if loop-seals in the cold leg were considered in that analysis.	92-94
Member Stetkar wanted to know under what conditions pilot operated safety relief valve (POSRV) discharge is directed to the containment atmosphere and not to the In-containment Refueling Water Storage Tank (IRWST) (normal discharge location), and why. KHNP clarified that IRWST is too small to maintain hydrogen concentration below flammability limits, hence for beyond-design basis accident (BDBA) conditions the discharge is sent to the containment atmosphere, for DBAs it goes to the IRWST.	94-97
Member Stetkar asked for assurance that the motor-operated three-way valves, on each of the two lines from the POSRVs to the IRWST, will not be blocked or severely restrict flow. KHNP took this as a follow-up item.	97-99
Member Skillman noted that DCD Tier 1, Figure 2.4.1-2, showed an alternate path of POSRV discharge to the Reactor Drain Tank (RDT), and asked where in DCD the RDT was discussed. He questioned the non-seismic design of POSRV piping to IRWST. It was noted that RTD, a part of CVCS, was discussed in Tier 2 Chapter 9 (and Tier 1, Section 2.4.6). The alternate line from POSRVs to the RDT is a path for valve leakage.	99-102

Non-safety gas vent operation during plant startup and shutdown discharges to RDT.	
Mr. Kim restarted his presentation on overpressure protection of the RCS and steam generators. Member Stetkar inquired about the required shutdown cooling system alignment for low temperature overpressure protection (LTOP) condition. A discussion followed.	102-108 Slides 8-11
Mr. Kim addressed a question from the members raised at the April 20 ACRS meeting regarding ASME qualification of the POSRVs, and that they meet ASME Section III requirements. Chairman Ballinger asked about heat treatment for the spring material Alloy X750, a high-strength nickel-based alloy. He noted certain heat treatments for that material are very susceptible to cracking. This question was taken for future follow-up.	109-112
KHNP noted that Slide 9 needed to be revised.	114-115
Mr. Kim discussed the open items in NRC staff SER. Chairman Ballinger read a question from member Corradini (not present) regarding the LTOP transient over-pressure relief. He asked whether energy input assumptions were conservative, specifically, uncertainty in energy input and the relief valve settings. Response was held off.	116-119
Member Stetkar asked if POSRVs were rated for water relief (vice steam). The answer was “yes.” However, the answer to the same question for MSRVs was “no,” it is specified for steam release only.	119-120
Mr. Jongsoo Kim, KHNP, presented Section 5.2.3, RCPB Materials, including material specification and selection, fabrication and processing, and open items in staff SER. Upon member Powers’ question discussion took place regarding avoidance of problems the Doral Plant in Belgium suffered with the reactor vessel forgings. Improved ultrasonic testing found quarter-sized flaws throughout the vessel, and hydrogen dissolution into the steel and an insufficient annealing time was noted as a possible cause. Member Powers voiced that chemical control may not be sufficient for forgings with low-alloy steel. Use of state-of-the-art ultrasonic detection technique was noted as paramount. The applicant (Mr. Sisk) stated that their evaluation of operating experience would address the issue and staff took it as a follow-up item.	120-136 Slides 12-16
Regarding Slide 13, “Austenitic Stainless Steel,” upon Chairman Ballinger’s question on definition of L-grade and a specification of less than 0.065% reduced carbon content a discussion took place on base and welding materials (3rd bullet).	125-128 Slide 13
Upon member Stetkar’s question, Mr. Tae Han Kim, KEPCO E&C, stated that metallic insulation is used inside containment.	128-131

Mr. Kim presented fabrication and processing of metallic materials. Chairman Ballinger wanted to know how ASTM A 262 is used for an as-welded structure. Mr. Kim noted that (performance of) welding material is ensured by welding qualification. Cleanliness specification, and an open item related to ensuring non-sensitization of materials were discussed.	131-136 Slide 15
Mr. Kim presented Section 5.2.4, In-service Inspection and Testing, including inspectability and inspector qualification requirements. Member Stetkar asked about accessibility to the RV bottom head due to high radiation dose rates in that area. In response Mr. Kim noted their experience in Korea.	136-138 Slide 17
Mr. Minseok Kim, KHNP, presented Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection with a short discussion of identified and unidentified leakage.	138-139 Slide 18
Mr. Jongsoo Kim, KHNP, presented an introduction to Section 5.3, Reactor Vessel Materials, with a description of the regulatory and ASME requirements it meets. Mr. Seognam Choi then presented Section 5.3.2.1, Pressure-Temperature (P-T) Limits, for a vessel life of 60 years. He discussed the P-T limit curves, pressurizer thermal shock events, and upper-self energy up to end of life conditions.	139-142 Slides 19-23
Mr. Jongsoo Kim presented Section 5.3.3, Reactor Vessel Integrity. Regarding surveillance capsules removal schedule (Table 5.3-5), member Stetkar asked why capsules are not removed from symmetric locations around the reactor vessel (removal sequence being 217, 37, 224 and 323 degrees, with 217 degrees and 224 degrees being in close proximity). Also, two spare capsules are not located 180 degrees apart around the vessel as one would expect. KHNP took the question for future follow-up.	142-153 Slide 24
Mr. Kim presented Section 5.4.1, Reactor Coolant Pump (RCP). He responded to two questions members asked at the previous April briefing. These had to do with design provisions for inspecting the reactor coolant pump rotor, and details of RCP shaft seal system cooling by seal injection from CVCS and high-pressure cooler. Mr. Kim noted seal design was different from the existing plant due to improved motors. A long discussion followed member Stetkar's question regarding how long can the pump run with no component cooling water (CCW) with seal injection available before the operators must trip the pump to prevent seal damage, and what test has been performed to confirm that. This question was to be followed up the next day.	153-165 Slides 25-31
Member Sunseri noted reliability issues with US positive displacement pumps, the type used for the auxiliary charging pump. KHNP explained its use during hydrostatic testing and noted its diversity from the main centrifugal charging pumps.	165-168

Mr. Kim mentioned the pump seal test and discussed the motor, the flywheel and SER open items related to its fracture toughness with pending test results, operating experience, material yield strength, and flaw size acceptance criteria. Upon member Riccardella's question, Mr. Kim noted there was no key way in the flywheel.	168-171
Mr. Seognam Choi, KHNP, presented Section 5.4.2, Stream Generators (SGs). Chairman Ballinger described the San Onofre experience, where two steam generators showed different degradations in spite of being manufactured under the same specification. A long discussion followed, with Mr. Choi describing the improvements made to the OPR1000 design. It was noted that KHNP used the EPRI (TH) codes and the steam generators are the same as in Shin-Kori Unit 3.	171-180 Slides 32-34
Mr. Tae Han Kim, KEPCO E&C, presented Section 5.4.7, Shutdown Cooling System. Member Stetkar asked if there was a manual isolation valve on the suction side of the containment spray pump. He was concerned about aligning a shutdown cooling pump to replace a containment spray pump without closing the suction valve from the IRWST. He also asked how the shutdown cooling pumps receive the automatic start signal from containment spray or safety injection. Mr. Kim described the valve interlock.	180-186 Slides 35, 36
Mr. Seognam Choi, KHNP, presented Section 5.4.2, Pressurizer. Member Stetkar had a question related to the basis for shutdown cooling system valve failure positions, given the valves are motor operated. KHNP took this as a follow-up item.	186-190 Slides 37-39
Mr. Kim presented reactor coolant system high point vents in Tier 2, Section 5.4.12. Member Stetkar wanted to see a drawing that shows all lines going into the IRWST and spargers. He mentioned manual valve V1430 in the RCGVS discharge to the IRWST, manual valve V212 in the line from reactor vessel head to the RCGVS, and manual valve V2300 in the connection from the pressurizer to RCGVS, and asked if those valves have position indication in the main control room, and the valves' repositioning requirements based on plant operating status. He also noted the PRA implication of solenoid valves failing closed after battery power is exhausted. He asked why POSRV leakage detection is not warranted.	190-203 Slide 40
Mr. Seognam Choi presented RCS Component Supports describing briefly the reactor vessel, steam generator, and pressurizer supports. Mr. Minseok Kim presented the summary and current status.	203-205 Slide 41-43
Although no formal presentation was scheduled on Appendix 5A, "Evaluation of the APR1400 Design and Intersystem LOCA Accident Challenges," member Stetkar noted that the analysis was a comprehensive assessment of interfacing system LOCA pathways. However, he noted	205-210

that the auxiliary pressurizer spray line was not included, and asked for the reason. KHNP took this as a follow-up item. He also asked for the volume of the RDT and a reference was provided. He wanted KHNP to confirm the relief capacities of each main steam safety valve (5.3% of full power) and POSRV (3.5% of full power). This was also taken as a follow-up item.	
Chairman Ballinger asked for comments from the public. No comments came forth.	210-211
Chairman Ballinger asked for comments from the members. Member Stetkar noted that he was looking at integrated plant safety, how the design relates to success criteria and risk assessment, and how the risk assessment results feed back into the design. He asked for less emphasis on open RAIs and more emphasis on design technical information in future meetings. Member Riccardella noted that he was impressed by the incorporation of lessons learned from existing plants especially in the materials area.	211-213
Chairman Ballinger adjourned the meeting at 3:27 pm	214

DAY 2: September 22, 2016

SIGNIFICANT ISSUES	
Issue	Reference Pages in Transcript
Chairman Ballinger started the second day of the meeting by introducing the ACRS members present, and noting the purpose of the meeting. He clarified that questions asked by ACRS members at the SC meeting are questions from each individual member alone and should not be taken as Committee position. And that the ACRS communicates through its letters only. Also questions from members are not RAIs.	220-221
Jessica Umana, NRO, stated the NRC staff presentation on their review of Chapter 5 by introducing the presenters and providing a summary of staff review. Mr. John Budzynski, NRO, presented staff review of Section 5.2.2, Overpressure Protection System, including a status of open items. Member Stetkar asked for a clarification of a statement in staff SER that noted a secondary temperature 230 degrees Fahrenheit, 110 degrees C greater than the RCS and found it to be more conservative than the 100 degrees Fahrenheit, 37 degrees C difference allowed by the technical specifications. He asked if it feasible to have a secondary temperature 230 degrees higher than the LTOP temperature of the primary system.	222-227 234-239 Slides 1-10

Mr. Budzynski, NRO, presented staff review of Section 5.4.1, Reactor Coolant Pumps. He stated the pump and motor bearings were designed to withstand the loss of CCW for 30 minutes. Member Stetkar noted that for pump over-speed and flywheel analysis the applicant assumed a maximum LOCA break size of 4 inches after applying the leak-before-break (LBB) method. He asked if the flywheel would come apart if the break size is larger, like 12 inches. John Honcharik, NRO, responded by saying that bigger lines will probably be assumed to leak and not break. Mr. Matt Mitchell, NRO, noted LBB is used for the elimination of the dynamic effects of pipe rupture, and took the question for follow up.	227-234 Slides 11,12
Mr. Budzynski presented staff review of Section 5.4.7, Shutdown Cooling, including evaluation of gas accumulation. This resulted in an SER open item regarding NEI 09-10, in that it contains the most current gas management procedure, but was not referenced in the DCD. Member Rempe asked for elaboration on how staff found the KHNP response (dated March 17, 2016) to RAI 384-8100 (Question 05.04.07-3) on natural circulation (power-to-flow ratio test acceptance criterion) acceptable.	239-242 Slides 13,14
Mr. Budzynski addressed staff review of Section 5.4.12, RCS high point vents. Mr. Stetkar questioned a statement in staff SER regarding bumping the RCP to remove gas accumulation in SG tubes during a natural circulation event. Staff took this question for future follow up.	242-245 Slide 15
Mr. Darren Widrevitz, NRO, presented staff review of Section 5.2.3, reactor coolant pressure boundary materials.	245-246 Slide 16
Ms. Jessica Umana, NRO, presented staff review of Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection. A discussion on technical specification (TS) allowable values of identified and unidentified leakage noted a typo in the SER. The typo had to do with SG leakage stated as 150 gallons per minute. It should be per day.	246-248 Slide 17
Mr. John Honcharik, NRO, presented staff review of Section 5.4.1.1, Reactor Coolant Pump Flywheel Integrity. He discussed the questions NRC reviewer asked and SER open items resulting from that. One question had to do with a failure analysis for the hub of the flywheel that has a place for a key. In a related question yesterday, from member Riccardella, KHNP stated there was no key in the actual design (P171 of transcript).	249-252 Slides 18-20
Mr. Greg Makar, NRO, presented Sections 5.4.2.1 and 5.4.2.2 on SG materials, and SG program. He noted the alignment of TS bases for SG leakage values related to that used in the accident analyses.	252-254 Slide 21

Member Stetkar noted the applicant had established 900# (40% of RCS normal operating pressure) as design criteria for piping outside containment for systems like shutdown cooling system, containment spray and safety injection systems as a reason why interfacing system LOCA for these systems with the RCS did not need to be considered, and staff had accepted that in the SER. He asked if that reasoning was the only basis for staff acceptance of the position, and if that was a formal NRC regulatory position related to interfacing system LOCAs. He also noted a report titled PNNL 24783, “Expert Elicitation to Support Interfacing System Loss of Coolant Accident Modeling,” and asked if it was still consistent with the NRC regulatory position, research position and so forth on susceptibility to interfacing system LOCAs.	254-259
Chairman Ballinger asked for comments from the public. No comments were offered.	259-260
After going through the members for additional comments, Chairman Ballinger adjourned the meeting at 9:31 a.m.	260-261

Following key follow-up issues resulted from questions and issues the members raised for which a response was not available at the meeting:

KEY FOLLOW-UP ISSUES	
Key Issues	Reference Pages in Transcript
KHNP to address design elements that limit the temperature of the zero percent exceedance non-coincident wet bulb temperature for the essential service water system to 81 degrees F.	31-33
Why in Chapter 2 KHNP references the site boundary when they mean the EAB.	36-38
Identify buildings that need to be protected for ground water given the values of maximum design flood and ground water levels.	38-40
Address the meaning of “artificially large floods” in DCD Section 2.4.4.2.	40
Address why failures of downstream dams or other impoundments was not mentioned regarding impact on the essential cooling water supply, or ultimate heat sink (re: DCD Section 2.4.11).	40-41
Address why ground water chemistry is not mentioned in the DCD (re: DCD Section 2.4.12).	41-43

KHNP to address in Chapter 3 presentation APR1400 buildings that are in the scope of the certified design.	47-50
Clarify statement in Section 2.1.1 that states a highway, railroad, or waterway may traverse the exclusion area but not close enough to the facility to interfere with normal operations.	53-54
Why hurricanes were not considered in SER Section 2.2.3.3, which considers tornado missile and pressure effects.	66-71
Inform ACRS staff if there was any change to staff SER Section 2.5 as a result of a recent RAI response post ACRS meeting.	84
Transients that are considered in 60 year design life analysis of RCS. Was RCP reverse rotation considered? Provide reverse flow delta T versus time that are included in the analysis (fatigue) in compliance with the Code.	87-89
If loop-seals in the cold leg were considered in the natural circulation cooling analysis.	93
Provide assurance that the motor-operated three-way valves, on each of the two lines from the POSRVs to the IRWST, will not be blocked or severely restrict flow.	97-99
Why piping from the POSRVs to the IRWST is non-seismic?	99
Describe what heat treatment is applied to the POSRV spring material Alloy X750.	110-112
Provide needed modification to KHNP Chapter 5 Slide #9.	114-115
Regarding the LTOP transient over-pressure relief what uncertainty assumptions in energy input and relief valve settings were used?	117-118
NRO staff to address problems Doral Plant in Belgium suffered regarding the forgings in their pressure vessel.	124-125
Reason behind selecting the locations of the four sample capsules and spares inside the RV.	143-153
If there a manual valve in the line between the junction of valve 340 and the suction of the containment spray pump.	183
Provide the basis for shutdown cooling system valve failure positions given the valves are motor operated.	187-190
Provide a drawing that shows all of the lines into the IRWST and location of the spargers.	191

Do manual valves V1430, V212, and V2300 in the RCGVS have position indication in the main control room, and if the valves are repositioned based on plant operating status.	196-198
Why POSRV leakage detection is not warranted?	201-203
Reason for not including the auxiliary pressurizer spray line in intersystem LOCA analysis (App. 5A).	206-207
Confirm the relief capacities of the each main steam safety valve (5.3% of full power) and POSRV (3.5% of full power).	209-210
Clarify a statement in staff SER that noted a secondary temperature 230 degrees Fahrenheit, 110 degrees C greater than the RCS and found it to be more conservative than the 100 degrees Fahrenheit, 37 degrees C difference allowed by the technical specifications.	225-227
Staff to elaborate on their basis for accepting the KHNP response (dated March 17, 2016) to RAI 384-8100 (Question 05.04.07-3) on natural circulation (power-to-flow ratio test acceptance criterion).	240-242
Clarify statement in staff SER regarding bumping the RCP to remove gas accumulation in SG tubes during a natural circulation event.	242-245
Member Stetkar's question on the basis for staff acceptance of KHNP position for not considering inter-system LOCAs for certain system piping and if an old PNNL report was still pertinent regarding NRC position.	254-259

Documents provided to the Subcommittee (CDs provided by KHNP)

1. APR1400 Design Description
2. APR1400 Design Control Document (Tier 1 and 2-Proprietary & Security Related Information)
3. APR1400 Topical Reports (Internal Use Only-Proprietary)
4. APR1400 Technical Reports (Proprietary)
5. System 80 Design DCD and SER
6. Staff SERs on Chapters 2 and 5

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
APR 1400 Subcommittee Meeting

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Wednesday, September 21, 2016

Work Order No.: NRC-2629 Pages 1-203

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

APR1400 SUBCOMMITTEE

+ + + + +

WEDNESDAY

SEPTEMBER 21, 2016

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Ronald G. Ballinger, Chairman, presiding.

DISCLAIMER

UNITED STATES NUCLEAR REGULATORY COMMISSION'S
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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CHARLES H. BROWN, JR., Member

JOSE A. MARCH-LEUBA, Member

DANA A. POWERS, Member

JOY REMPE, Member

PETER C. RICCARDELLA, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Chairman

MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER L. BROWN

ALSO PRESENT:

TONY AHN, KHNP

JOHN BUDZYNSKI, NRO

BOB CALDWELL, NRO

LUISSETTE CANDELARIO, NRO

SEOGNAM CHOI, KHNP

WOOCHONG CHON, KEPCO NF

JEFF CIOCCO, NRO

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GYUN DO JEONG, KHNP

HYGOK JEONG, KEPCO E&C

JONGSOO KIM, KHNP

JUNGHO KIM, KHNP

MINSEOK KIM, KHNP

TAE HAN KIM, KEPCO E&C

YUNHO KIM, KHNP

KWANG-HOON KOH, KHNP

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TAE SUN RO, KEPCO E&C

JAMES ROSS, KHNP/AECOM

TARUN ROY, NRO

ROB SISK, Westinghouse

DUK BIN SONG, KEPCO E&C

JAMES STECKEL, NRO

ANGELO STUBBS, NRO

JEONG KWAN SUH, KHNP

SESHAGIRI TAMMARA, NRO

VAUGHN THOMAS, NRO

JESSICA UMANA, NRO

WILLIAM WARD, NRO

JASON WHITE, NRO

DAN WIDREVITZ, NRO

SEOK JEONG YUNE, KEPCO E&C

*Present via telephone

A G E N D A

P R O C E E D I N G S

8:30 a.m.

CHAIRMAN BALLINGER: The meeting will come to order. This is a meeting of the APR1400 Subcommittee on the Advisory -- of the Advisory Committee of Reactor Safeguards.

I'm Ron Ballinger, Chairman of the APR1400 Subcommittee. ACRS Members present are hopefully Pete Riccardella, Dick Skillman, Dana Powers, Matt Sunseri, John Stetkar, Jose March-Leuba, Joy Rempe, Chris Brown, not Chris Brown --

MEMBER BROWN: Charlie Brown.

CHAIRMAN BALLINGER: Charlie Brown, and whoever else might show up halfway through this.

The purpose of today's meeting is for the Subcommittee to receive briefings from -- by the way, Chris Brown is the Designated Federal Official.

The purpose of today's meeting is for the Subcommittee to receive briefings from KHNP regarding their Design Certification Application, and NRC Staff regarding their review of the Safety Evaluation, specifically Chapters 2, with the exception of 2.3, I think, Site Characteristics, and 5, Reactor Coolant and Connected Systems.

During our Subcommittee meeting on April 20th and 21st we received an extensive briefing from KHNP on the overall design and various aspects of their application that is of interest to the ACRS Members.

The rules for participation in today's meeting were announced in the Federal Register on September 7th, 2016. The meeting was announced as an open/closed to public meeting. This means that we can close the meeting, as needed, to protect information proprietary to KHNP or its vendors, and you'll have to remind us if that's the case; if we ask a question which is -- goes into proprietary information, we can close the meeting. No requests for anybody making a statement to the Subcommittee has been received from the public.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice. Therefore, we request that participants in this meeting use the microphones, including myself, located throughout the meeting room when addressing the Subcommittee, and please identify yourself for the transcriber. Participants should first identify themselves, right, and speak with sufficient clarity and volume so that they can

be readily heard.

While we have a bridge line established for interested members of the public to listen in, I would advise members of the public that are listening in to put your device on mute because we run the risk of getting circular feedback which is just -- makes it tough. The bridge line number and password were published in the agenda posted on the NRC website.

The public will have an opportunity to make a statement or provide comments at a designated time towards the end of the meeting. We request that the meeting attendees and participants silence your cell phones and other electronic devices.

I now invite Jeff Ciocco, NPO -- oops, who's going to do the introduction? Oh, Bill Ward, to introduce the presenters and start the briefing.

MR. WARD: Good morning. My name is Bill Ward. I'm the Acting Branch Chief of Licensing Branch 2, which is the branch managing the review of the APR1400 design.

I'd like to thank everybody for being here. We look forward to many more meetings. This is the first of all the Phase 2, Phase 3 ACRS

meetings, Subcommittee meetings that we're having. Over the next several months we intend to get through all the chapters, and we look forward to many more.

MR. CIOCCO: Thank you, Bill. My name is Jeff Ciocco. I'm the Lead Project Manager for the APR1400 Design Certification Project. I also want to thank the Subcommittee Chair and your staff, which we've worked very closely over the past year and a half to establish this meeting, and as Bill said, the following Phase 3 meetings on the APR1400 Design Certification Project.

This meeting is a very significant milestone for us in the review process in that we have begun completing our Phase 2 Safety Evaluation Reports with Open Items. Today is a small, albeit important sampling of the 22 Safety Evaluations that we're going to be presenting and you'll be reviewing over the next seven or nine months.

Back in April, I gave a project overview presentation to the ACRS, and KHNP and KEPCO gave a design overview, so I don't want to repeat everything that I said back then, but just briefly regarding this application.

We received the APR1400 Design

Certification Application in December of 2014. We docketed the application in March of 2015, so right now we are about a year and a half into our 42-month review schedule leading to the potential issuance of the final Safety Evaluation Report and rulemaking.

This morning in KHNP's presentation on the design overview they have included the overall project schedule, so if you have any questions for me at that time, that would be a great point.

And, finally, I certainly want to thank our technical staff and project managers here today who are going to present and answer your questions on Chapters 2 and 5. They are the engines of our review. So thank you very much.

CHAIRMAN BALLINGER: KHNP?

MR. Y. KIM: I am Yunho Kim. I am technical project manager for APR1400, and we have just, of course, go through NRC's review for our (inaudible) document, so I think that we are ready to go through your review, so we will look forward to communicating with you.

MEMBER STETKAR: If you all could just either pull the microphones closer and make sure you speak up because all of our transcript comes

from the microphones, so we need to make sure that you're loud enough so that we hear you on the transcript.

MR. Y. KIM: Yes.

MEMBER STETKAR: So be careful about that, please.

MR. Y. KIM: Yes, thank you. So I think that I am going to present my material?

CHAIRMAN BALLINGER: Yes.

MR. Y. KIM: Yes. Today, this morning I'm going to present overview of APR1400 DC project.

The application of Design Certification was done by KEPCO and KHNP. KEPCO and KHNP's co-applicant project. KEPCO is KHNP's mother company, and KHNP is the only Korean utility company in Korea, and also technical management, but this project was done by KHNP. So just to let you know that our applicant information.

Today, I'm going to present the first introduction as introduction. I'm going to present the (inaudible) overview of APR1400, and nuclear power plant in Korea. And, also, I will present history. After that, I am going to present design features and the general arrangement for APR1400.

And also (inaudible) docketing our licensing material at December 2, 2014. We go through the review of NRC Staff, and I will summarize that review status. And, finally, I will summarize my presentation.

This list of materials that we presented last April, this year, for your understanding of APR1400. That is from I think the major design area of APR1400 was presented at that time. So going over the material overview of this project, this is same material that was done before, last April, so April you are interest in our design. You can see this material. I understand that this material is available, too.

First, let me introduce nuclear power plant in Korea. Currently, we have 24 units is in operation. Major site is that we have the Hanul, west side of Korea, and east side of Korea there are three major sites; first is Hanul site, second one is Wolsong site, and third site, Kori site. And we are now commissioning one unit, and the third unit is under construction, and six unit is planned. So one unit in commissioning, that's Shinkori 3. This is the reference plant for APR1400. We believe that this plant is commercial

operation within a month. And three unit are under construction, and Shinkori 4 will be commissioning within next year. And Shin-Hanul 1 and 2, this is also APR1400, that is under construction. And all the remaining planning site is that where construction will be the APR1400 (inaudible) with more advanced design. We call the APR Plus Design, to be constructed, yes. So this is the current -- our nuclear power plant.

Let me add that the Wolsong 1, 2, 3, and 4 are the CANDU type. This is slightly different from most of our pressurized water nuclear design.

MEMBER SKILLMAN: You know, for the planning units, the six units for 8600 megawatts, that is 200 megawatts more than six times 1400.

MR. Y. KIM: For the --

MEMBER SKILLMAN: So these are advanced units that are a higher power than 1400?

MR. Y. KIM: Right. Our current design APR1400 megawatt electricity, for APR Plus Design has more (inaudible) these 1500, so 100 more.

MEMBER SKILLMAN: Is the 1500 megawatt electric design the same basic design with some enhancements?

MR. Y. KIM: Basic design may be similar plus some enhanced design is incorporated in there like passive cooling, that kind of thing is incorporated in APR Plus Design. Also, let me explain that like the instrument in core, in-core instrument is from down to top, but our new design comes from top to down, so that comes with (inaudible) incorporate, and also containment spray is -- containment cooling is done with a passive system, that kind of thing is incorporated -- will be incorporated in our APR Plus Design.

MEMBER SKILLMAN: Thank you.

MR. Y. KIM: Next I'm going to present our project organization. As I said earlier, KEPCO, KHNP is co-applicant on this project. KEPCO's role is that they provide funding for this project and that they determine the policy decision for this project. And KHNP is manager for this project in terms of technical management or quality assurance, that kind of thing is done by KHNP. So project management was done by KHNP, and the major design is done by Korea domestic company like fuel and core design is done by KEPCO Nuclear Fuel, we call KEPCO NF. NSSS design is done by KEPCO E&C. That is the -- and the balance of plant design is done by

KEPCO E&CAE. AE stands for architectural engineering, SD stands for system design. And the major component like reactor vessel, steam generator, pressurizer, that kind of design is done by (inaudible).

Also, we have overseas consultation from Westinghouse, and AECOM. Westinghouse owns the combustion engineering design features our APR14 design is based on, so we have mutual agreement, technology agreement with Westinghouse, so we share information. And from AECOM we have got licensing support. We have -- there are many licensing activities between NRC Staff and KHNP, so AECOM is supporter. And the thermal hydraulic test is done by KAERI. KAERI is Korea's major nuclear research and development facility, so we did TH test from KAERI like fluidic device. We have in the safety injection tank we have a fluidic device, so for that fluidic device some hydro tests, that is done by KAERI. This is the whole our project organization scheme.

Then let me brief our APR1400 Design Certification history and progress. APR1400 project for Design Certification goes back to 2009. KEPCO and KHNP determined to apply for Design

Certification. We had long pre-application review period from 2010 to 2014. During the pre-application period, KHNP submitted five topical report which are quality assurance program, PLUS7 Fuel design, critical heat flux correlation, fluidic device design, and the large break LOCA methodology. In large break LOCA we applied the best estimate.

Among these five topical report Staff issued the Draft Safety Evaluation Report for Quality Assurance Program, fluidic device, and critical heat flux topical report. We have NRC Staff issued three SER reports for these three topical reports.

For the remaining topical report, KHNP expects they will issue Draft SER once KHNP addresses that concern. I believe those will be also under your review.

In 2014, KHNP submitted design control document and the 51 technical report. We believe we have much complete information for APR1400 Design Certification. KHNP provided additional 19 technical report (inaudible) which was done by March 2015. Since then, KHNP has more than 2000 RAIs and several hundreds of action items which

came from conference meeting or conference call, face-to-face meeting. KHNP believes most of them are addressed except for some open items that was described in our SER. KHNP believe these open items will be also addressed in the near future. This is the history and this picture was done at the time of we submit our application.

Next, let me brief development history of APR1400. APR1400 design was developed, development that go back to 1992. We developed this design almost 10 years. Our design is based on combustion engineering design so at that time in early 1990 we had the license agreement with -- at that time we call it ABB-CE. So our design is based on Palo Verde design in terms of NSSS design, and from core design we have anode number two C design. So based on this combustion engineering design we standardized our nuclear power plant design. At that time we call it Korean Standard Nuclear Power Plant, but we change it. The plant name for (inaudible) so we call it OPR 1000.

OPR 1000 has very much operating experience. Hanbit 3 and 4, Hanbit 5 and 6, Hanul 3 and 4, Hanul 5 and 6. Those are the designed with OPR design in operation, and also we have some

improved OPR 1000. That is just -- we call it improved, means some kind of economy benefit. We just reconfiguration some of the systems so we call it improved OPR 1000, but we think that that is the same (inaudible) of the OPR 1000. Shin-Kori 1 and 2 is that we call it improved OPR 1000.

For this operation of OPR 1000 we have very much experience in our design, so based on this experience, accumulated experience we developed APR1400. So APR1400, also we call global marketing, we just (inaudible) for the URD document and System 8 Plus design, and for NRC project, we apply the latest code and standard. That is the basis of our APR1400 development.

Then next let me introduce our APR1400 (inaudible) or just our current status of our APR1400 design. APR1400 design (inaudible) Shin-Kori 3 employees are so -- Shin-Kori 3 is almost -- we put in the pure and we right now, as of today, right now we do the hot function test so we are almost done all the test, and we are waiting for our party's approval for commercial operation. So we think that APR1400 is a finished and complete design.

In the commercial operation is

scheduled October 2016, as I mentioned earlier. And also we have APR1400 contract with UAE in Barakah, so in Barakah also design same as Shin-Kori 3 and 4, and it is same as APR1400. This Barakah project, construction schedule is on schedule, so they will be also into operation next year, so that may be our current schedule. And Korea also, for Barakah project we have -- KHNP have contract with ENEC and we are going to support, all the operational people, maintenance people will be provided for operation of the Barakah project.

MEMBER REMPE: Excuse me. How many years did it take to construct Barakah?

MR. Y. KIM: Barakah, we have around 15 months.

MEMBER REMPE: Fifteen months.

MR. Y. KIM: Yes.

MEMBER REMPE: Thank you.

MR. Y. KIM: The APR1400 design feature, this slide is the same as the one we are going to present in Chapter 5 presentation, so I just want to just touch it very briefly, and then I will hold it to more detailed explanation in Chapter 5 presentation.

Basically, our APR1400, the lifetime

you call it 60 years for major (inaudible). And we -- APR1400 is a two-loop plant, two steam generator, RCP, cold leg, and the one pressurizer. And let me touch -- let me skip this slide.

So basic approach for APR1400 design for NRC project, design certification project, you are going to retain (inaudible) as much as possible we can. So that is just to take the benefits of proven safety and performance. Also for design certification we have U.S. NRC guideline that is effective on August of 2014. This is six months before we are out of docketing date, so we just apply all the regulation guide effective on August 2014.

Let me add some APR1400 design feature. That is a slide from Shin-Kori 3 and 4. One is that we -- the station blackout coping capability. With (inaudible) that capability, we applied gas turbine generator for alternate current source, and also we increased the battery capacity. That will increase our (inaudible) and also our (inaudible) capability. Also, we applied FLEX implementation such as standard pressure for water level measurement, and spent fuel spray, and the water hookup to supply cooling water from outside. Also,

we add hookup for the primary side and the secondary side, so that kind of capability we kept in APR1400. And also --

MEMBER SKILLMAN: What is the basis of the 16-hour battery? What is the load?

MR. Y. KIM: Sixteen hour battery, that is based on -- plus in Shin-Kori 3 and 4 we have 8-hour battery capacity, but in terms of we are going to increase our (inaudible), and also we have some kind of -- based on the NRC RG 1.155, there is some guidance for nuclear power plant SBO coping time. So at that time, that is based on the emergency alternate current power system configuration, and also the reliability of emergency diesel generator, and also the frequence of loss of offsite power. That kind of basis on these category, that is our design is classified as P3 category, and the P3 category requires 16 hours.

MEMBER SKILLMAN: Thank you.

MEMBER STETKAR: We'll get into this once we've reviewed Chapter 8, but in fact, on the DCD they have four batteries. And I believe that trains A and B in the certified design are 8-hour batteries because they can be recharged from the alternate AC generator. Trains C and D are 16-hour

batteries because they cannot be recharged from the alternate AC generator, at least not easily. Is that correct?

MR. Y. KIM: Right.

MEMBER SKILLMAN: Thank you.

CHAIRMAN BALLINGER: Are any of these enhancements, if you will, being fed into the UAE design, or is the UAE design strictly Shin-Kori Unit 3?

MR. Y. KIM: Well, this is based on -- we just -- our objective is to apply for NRC design certification, so --

CHAIRMAN BALLINGER: Thank you.

MR. Y. KIM: -- we can say that. Like the aircraft impact is -- that is it depends on each country.

The second one is that we improve the tolerance part of beyond the design basis such as aircraft impact, and also loss of large area, and physical security requirement, so we incorporate all the U.S. requirement in our APR1400 design.

Also, one of the big issue is the GSI-191, as you know, so we did GSI-191 test for large break LOCA, and the upstream and the downstream test is done, and NRC Staff audited our facility,

so I think we believe GSI-191, we successfully completed.

Also, we have diverse protection system for common cause failure, so usually plant protection system is word for core protection, but we also have diverse reactor protection system. And for the seismic design, until Shin-Kori 3 and 4 design we applied the ASTM model. It's very crude, but for APR1400 design we applied finite element of this in our seismic design. This is, I think, a major difference from our Korean design, and for U.S. NRC APR1400 design.

And the next one is, let me explain about what is the difference between APR1400 System 80+. In terms of containment respects, System 80+ is spherical steel design and APR1400 is cylindrical pre-stressed concrete design. And in terms of thermal power there's a slight difference. System 80+ is 3931 and APR 1400 4000 megawatts thermal. And we --- in hot leg temperature, we already decreased our hot leg temperature. And also, in terms of safety injection system we added a fluidic device in our safety injection system. I think we believe that with this fluidic device we can enhance for small break LOCA capability, coping

capability. So just we will use (inaudible) the safety injection system.

And in terms of over-pressure protection system, System 80+ is equipped with pressure safety valve and two safety pressure system, but we incorporated this design feature into -- in one system. We call it POSRV, so POSRV does (inaudible) capability also depressurization system capability in our POSRV.

And in terms of reactor vessel overhead structure for our maintenance, maintenance convenience we applied integrator head assembly, so easily we can remove reactor vessel upper head system during the overhaul period. And also in terms in the severe accident we -- System 80+ is a cavity flooding system but we added power and in-vessel retention and reactor vessel cooling capability. For this design feature we can discuss in our separate chapter review time.

For general arrangement, this is -- our design certification is a single unit basis, and single unit base for reactor containment is surrounded by auxiliary building. In auxiliary building we have two EDG inside, two EDGs outside, and we have turbine building, and one compounded.

This is the general arrangement for APR1400. And this shows the primary reactor coolant system. Most of this design is the same as the System 80+ except for the POSRV, as I explained earlier, that this has combined capability of safety relief and depressurization capability.

And our general equipment arrangement in Aux building is we have four quadrant, so we have some high function in one quadrant, but that does not protect, and that doesn't impact that safety injection pump is located for each quadrant, and shutdown pump, and the containment spray pump is located for each quadrant, and the component cooling pump is located for each quadrant. So we have this four quadrant design feature in our APR1400.

Then next let me explain about our APR1400 design certification review status. NRC is planning to have complete this design review within 42 month schedule, so Phase 1, the target date was established by NRC, and Phase 1 is completed January this year. We believe they have successfully completed Phase 1, and we are now at the stage, final stage of Phase 2 stage, and the schedule is November 2016. So that maybe depend on

each chapter progress, but we believe Chapter 2 and Chapter 5 is now reached its final stages so we are on the track. And Phase 3 and 4 is also, Chapter 2 and 5 is underway, Phase 3 is underway, and Phase 4 is underway, so NRC plan to complete Phase 4 review at end of 2017. So KHNP tried to support and meet this target schedule. We will work very hard to support this schedule.

So let me summarize for this APR1400 project. APR1400 adopt proven technology from operation of OPR 1000, and so APR1400 use safety code for System 80+. So in terms of technical issue we have based on -- we have accumulated experience and also we have certified design code. And in Korea, APR1400 standard design was approved by Korean regulatory authority in 2002, and as I mentioned earlier, Shin-Kori 3 will be commercial operation soon, and Barakah also soon, so I think that APR1400 is essentially very complete design. So that is the message I want to give you this morning. Thank you.

CHAIRMAN BALLINGER: Thank you. Questions from the Committee Members? We're already a little bit behind, so we need to spend the time that's required but be mindful, please, of the

time.

MEMBER REMPE: Mr. Subcommittee Chairman, I think --

CHAIRMAN BALLINGER: My schedule shows 8:40 to 9. Oh, pardon my dyslexia.

(Laughter)

MEMBER STETKAR: We'll take care of the clock and the microphones. You just run the meeting.

CHAIRMAN BALLINGER: Can you change my diapers?

(Laughter)

MR. SISK: I think we're through with the overview, if there's any questions. Otherwise, we can move into Chapter 2 very quickly, so we are mindful of time, as well. So let's switch out.

CHAIRMAN BALLINGER: Yes.

MR. SISK: Okay. Are we ready, Mr. Chairman?

CHAIRMAN BALLINGER: Yes.

MR. SISK: I am Rob Sisk, Westinghouse. I'm just going to briefly, quickly introduce Mr. Kwanghoon Koh. He will lead us through the Chapter 2 presentation. And with that, I'll turn it over to Mr. Koh.

MR. KOH: Let me start. Good morning, ladies and gentlemen. My name is Kwanghoon Koh. I working for KEPCO E&C. I'm a civil engineering specialist. For today's presentation, civil engineering and the structural part will be presented by me, Kwang Koh, and the mechanical and nuclear engineering part will be presented by Mr. Dong-Su Lee. Now, I am going to present our work results entitled APR1400 Chapter 2, Site Characteristics. Next slide.

Presentation of Chapter 2, Site Characteristics and the summary of this presentation will be given. The documents for APR1400 are as follows; APR1400 DCD design control document, Tier 1 and Tier 2. After the meeting, many RAIs, number respond 33, number of open items is zero.

The scope of this Chapter 2 is as follows. Chapter 2 provides the five requirements for the APR1400 design, including geological, seismological, hydrological, and meteorological characteristics. The COLA is to come from -- to provide the site specific qualification.

This slide is for site characteristics. APR1400 is designed on the basis of a set of rules,

to include a range of potential nuclear power plant sites in the United States. Table 20-1 is a summary identifying the APR1400 design.

This slide is for (inaudible). Section 2.1 is geography and tomography, and Section 2.2 is called industrial transportation and military facilities. These two sections will provide the (inaudible) descriptions here. List of criteria for Chapter 2 is provided in the attachment.

Chapter 2.3 is for meteorology. Some (inaudible) snow, winter precipitation (inaudible). These are based on HMR, Hydro Meteorological Report and the SEG 7.

There are three wind related parameters, wind, tornado, and hurricane. These parameters are based on ASCE 7, tornado parameters are like these. These parameters are based on Reg. Guide 1.276, and the hurricane parameters are like this, and it is based on Reg. Guide 1.221.

These three parameters are found in most Continental weather. HVAC temperature and beyond design temperature and.

CHAIRMAN BALLINGER: I'm reminded by Chris that KHNP is presenting Chapter 2.3, but the Staff will not be presenting on Chapter 2.3, to

my knowledge.

MR. CIOCCO: That's correct, not today. We will do that at the October 4th meeting.

CHAIRMAN BALLINGER: Right.

MR. CIOCCO: Our Senior Meteorologist wasn't available, but you have the Safety Evaluation which has all of our 2.3.

CHAIRMAN BALLINGER: Yes.

MR. CIOCCO: Thank you.

CHAIRMAN BALLINGER: Thank you.

MR. D. LEE: Thank you, sir. My name is Dong-Su Lee, who is the second presenter.

As you can see in this slide, the HVAC design temperatures here. The HVAC systems are designed applying outdoor air temperature specified in the FURD. The outdoor air temperatures are categorized into three conditions as is shown in this slide; 5 percent exceedance value, 1 percent exceedance value, and zero percent. The 5 percent annual exceedance values are used to design (inaudible) design, and 1 percent is non-safety related design, and the zero percent are used on safety-related design. Let's move onto next page.

In this page, we had correction on this slide due to some typos from the version we provide

previously. In beyond design temperatures for power are categorized into two parts. First part is 5 percent exceedance value for the circulating water system. Second one is zero percent exceedance value for essential service water system. Let's move on to the next page.

MEMBER STETKAR: Can we stop right there?

MR. D. LEE: Yes.

MEMBER STETKAR: The 81 degrees Fahrenheit, zero percent exceedance non-coincident wet bulb temperature for the essential service water system is very low for the United States.

MR. D. LEE: We know that.

MEMBER STETKAR: Okay. That's -- and I'll ask the Staff during our next meeting why they accept that. But if I look at several other designs that have been certified in the United States, that temperature tends to be in the 86 to 88 or 89 degree range. I have an anecdote from the Staff SER on another certified design that said 21 percent of the weather stations in the United States had a zero percent exceedance non-coincident wet bulb temperature that exceeded 86 degrees Fahrenheit, which is 5 degrees higher than your's. So from the

perspective of your design, what in your design limits that temperature to being so low? What specific elements of your design? Is it your design specifications for your essential service water cooling towers? Is it temperature limits on your component cooling water system? Is it chilled water systems for HVAC? What elements of your design limit that temperature?

MR. D. LEE: Okay, thank you. Let me first introduce that the key review items --

MEMBER STETKAR: That's -- I'm asking you -- I've read the RAIs. I'm asking you what elements of your design limit that temperature? I know that your RAI response says that it was accepted for the ABWR. The ABWR combined license for the South Texas plant, they had to increase it to 88.3 degrees Fahrenheit for that site. That is an extreme site, I will admit it, but I'm asking you what elements of your plant design, not politics, what elements of your plant design limit that temperature to 81 degrees Fahrenheit? Why can it not be something like 87 or 88 degrees Fahrenheit? And if you can't answer that, that's fine. We can take the answers at a later meeting, especially because we're covering Chapter 2.3 in

another Subcommittee meeting.

MR. D. LEE: Thank you.

MEMBER STETKAR: But I'd like to know that clearly in terms of where your thermal-hydraulics limit that temperature.

MR. SISK: So let me kind of answer that a little, give you as precise answer, we can get back --

MEMBER STETKAR: Rob, turn your microphone on.

MR. SISK: This is Rob Sisk, Westinghouse. Thank you, John.

We need to go back and get you a -- look at that question specifically.

MEMBER STETKAR: Yes.

MR. SISK: As we noted, you'll be talking to 2.3 at a later time. If we don't have an answer later today, we'll get back to you by that time.

MEMBER STETKAR: I thought you were going to have to do that. I just wanted to get it on the record so you knew next time we need a list to come back with.

MR. SISK: We have it noted, and thank you.

MEMBER STETKAR: Thank you.

MR. BROWN: Maitri and I will track that as an action item.

MR. D. LEE: Let me start. The atmospheric dispersion factors were determined for the dose recovery during normal operation, and for the radiological consequence analysis for (inaudible) conditions. And on-site atmospheric dispersion factors are used for the MCR/TSC habitability analysis. The regulatory basis are as follows.

In DCD 202, Table 20-1, Bounding Atmospheric Dispersion Factors and Dispersion Factors are presented. Once the on-site atmospheric dispersion factors for MCR/TSC habitability analysis are present in DCD Table 23-2 through 23-12, and the input variables including post (inaudible) and intake locations used for the on-site (inaudible) are shown in the Table 23-13 and Figure 23-1. Let's move on to the next page.

In the next page this slide shows the design features for the atmospheric dispersion factors. The long-term factors (inaudible) were selected to bound the most (inaudible) site, and the short-term atmospheric dispersion factor at EAB

was obtained from EPRI-URD. For the on-site atmospheric dispersion factors have been calculated based on the selected materials (inaudible) to limit among six U.S. sites. And the two enveloped most on-site (inaudible) U.S. sites, the sufficient margin was applied. Lastly, ARCON96 code was used.

MEMBER SKILLMAN: Let me ask this question, please. As I understand it, the site that you chose is Prairie Island. Is that the most limiting meteorological site?

MR. D. LEE: I will give you answer for you. To address the essence of the single sites providing limiting (inaudible) values for the APR1400(inaudible) geometrics. The onsite χ/Q values are analyzed using 5-year Prairie Island (inaudible) data as you mentioned, and the resulting χ/Q values are increased by 50 percent such that more onsite χ/Q values become bounding for the U.S. site meteorological data.

MEMBER SKILLMAN: Thank you.

MR. D. LEE: On the next page, this slide shows the (inaudible) set perimeter values for APR1400 compared to the corresponding site characteristics, and at the other design site perimeter values. It is shown in this table

atmospheric dispersion factor and (inaudible) factor taken into account for the APR1400 could bound other sites in U.S. Therefore, these sites perimeters values are reasonable for use in off-site dose assessment.

MEMBER REMPE: Before you leave this topic, I would normally ask this question of the Staff if they were going to discuss Section 2.3 today, but they're not here to discuss that today, so I'm going to try and phrase it as a question to you, and I'm hoping the Staff will speak up and clarify. But in their Draft SER on page 245 they said, sometimes Chapter 2 references the site boundary when they mean the EAB. Are you aware of the Staff's comment in their Draft SER on page 245? And do you know why the Staff provided that comment in the Draft SER on that page, because I'm wondering if this comment means that they had a concern with Section 2.3, because they kind of imply that the way that Section 2 was written, that there might be an error. And I just want to know what was going on. And so could somebody clarify that statement and what it means, and did they discuss that with you?

MR. CIOCCO: I don't think we have our

meteorologist here.

MEMBER REMPE: Well, again, are you aware of that statement in the Staff's --

MR. CIOCCO: Excuse me. We do have somebody here.

MEMBER REMPE: Okay, good.

MR. MAZAIKA: This is Mike Mazaika from the Hydrology-Meteorology Branch.

There's sometimes a terminology issue that we were questioning where the site boundary and the EAB are co-located, and in some cases that they are separate, and they're used for two different analyses. The site boundary is often used for the long-term χ/Q (inaudible) impact analyses, the EAB is used for design-basis accidents.

MEMBER REMPE: Was this discussed with KHNP, and was it clarified so that you feel comfortable that things are correct now?

MR. MAZAIKA: I would have to go back and look at the RAI response. I don't have that answer right now.

MEMBER REMPE: Okay. Yes, we'd like to -- I would -- the sentence didn't reference an RAI, and so I just would like some clarification of that

sentence, and to understand that things are okay now.

MR. MAZAIKA: Okay. We'll look at that.

MEMBER REMPE: Thank you.

CHAIRMAN BALLINGER: It's on the Action Item list.

MR. KOH: Section 2.4 describes hydraulic engineering. Groundwater and flood elevation (inaudible). Groundwater elevation is two feet below plant grade, maximum level for flood is one feet below plant grade. These parameters are based on EPRI, URD, ARWR, maximum rainfall precipitation rate (inaudible) hourly, 90 for each hour, and the short term 3.2 inches in (inaudible). These parameters are based on hydro meteorological report and the EPRI value.

MEMBER STETKAR: Mr. Koh, before we leave that slide, in Table 2.0-1, it's -- I'm confused about the words that are used throughout the DCD. So, for example, in Table 2.0-1, it actually says maximum elevation of groundwater 0.61 meters, 2 feet below plant grade in the vicinity of the SSCs important to safety. In other places I read, safety-related elevations, and you have in Chapter 19 and Chapter 17 a list of SSCs that are

risk-significant. So what are the buildings that have SSCs important to safety? Are they only safety-related? Are they risk-significant SSCs? What are those buildings, because it's really important for me to understand that?

MR. KOH: The building -- the SSC important to safety building partially it is we think that the(inaudible) and the related -- safety-related buildings, and also some buildings under -- around the -- the buildings around the safety-related building potentially also are including.

MEMBER STETKAR: I didn't quite fully understand that, because you said safety-related buildings and other buildings around there. I'm really interested to understand if I'm going to build this plant what buildings do I need to assure that the groundwater is more than two feet below grade and the flood is more than one foot below grade?

For example, turbine generator building, is that yes or no? A simple question, yes or no? I will tell you that the turbine generator building does contain SSCs that are risk-significant according to Chapter 19 and Chapter 17,

which you're well aware of, as do many other structures. You can answer at a later date, but I want to be very, very clear on this so that there's no question.

MR. KOH: Okay.

MEMBER STETKAR: Don't change the -- oh, unless you have an answer.

MR. KOH: I will provide the information later.

MEMBER STETKAR: Okay. Then on the topic of -- because this is all hydrology. In Section 2.4.4.2 you make a statement that says site-specific information related to the effect of dam failures on the site includes how the analytical methods that are presented are applicable to artificially large floods with appropriately acceptable coefficients and flood waves through reservoirs downstream of failures. What do you mean by artificially large floods?

MR. SISK: You're talking now about temporary flooding of the site?

MEMBER STETKAR: It's on the record.

MR. SISK: Thank you.

MEMBER STETKAR: The -- in Section 2.4.11 where you discuss low water considerations,

you specifically mention "site-specific information related to low water considerations includes low flow in rivers and streams, low water resulting from surges, seiches, or tsunamis, historical low water, future controls, plant requirements, and heat sink dependability requirements." You don't mention explicitly the fact that as a COL applicant, I should also evaluate failures of downstream or other impoundments for my essential cooling water supply, or my ultimate heat sink, if I so desire, decide to construct my plant that way, because part of the ultimate heat sink is my responsibility as a COL applicant. So I was curious why you didn't provide any notion of evaluating failures of water impoundments or downstream dams if I'm on a river, but it could be an impoundment of a cooling water reservoir, for example. We have reviewed one site in the United States that has a very, very large cooling water reservoir with an earthen dyke around it. You might call it a dam, it's a dyke, whatever. And failures of that structure can cause problems for them. So I'll put that on the record just to see what your response would be.

And in Section 2.4.12 regarding

groundwater, the discussion in that section relates to hydrostatic loading on structures, and I understand that. But I couldn't find anything in the DCD, at least quickly, that relates to groundwater chemistry. In other words, do you have limits on the chemistry of the groundwater in terms of aggressiveness, chlorides, for example, because those may compromise below-grade structures. And, obviously, since the groundwater can be only two feet below grade, a substantial portion of the subsurface structures could be exposed to the groundwater. So I was curious, if you do specify some sort of chemistry when it's for the groundwater where are they specified in the DCD?

MR. KOH: I think that you groundwater chemical problem related to problems --- (inaudible) different from site to site, so --

MEMBER STETKAR: That's absolutely --

MR. KOH: --- (inaudible).

MEMBER STETKAR: Absolutely true but if, for example, depending on your specifications for the type of concrete that you're going to use in your buildings it may be more or less susceptible to groundwater chemistry. So as a COL applicant, if there are specific limits on the groundwater

chemistry that I need to be aware of, I would think that they should be specified in the DCD. If there's no limits, you know, I could probably put it in an acid bath. So where do you specify those limits?

MR. SISK: We have it --

MEMBER STETKAR: If you do have limits, they should be specified such that the COL applicant must confirm that their groundwater chemistry, you know, is better than those limits. If there are no limits, that's fine, but it's rare that that's actually the case.

MR. SISK: Thank you for the question. This is Rob Sisk, Westinghouse. Thank you for the question. We will -- we have noted the question down. I will need to evaluate it and get back.

MEMBER STETKAR: Thank you.

(Off record comments)

MR. KOH: Can I start again? Okay. Section 2.5 describes geology, seismology, and geotechnical engineering. SSE CSDRS and the HCRHRFSO are used for seismic design of structural and for (inaudible). SSE is based on 0.3g peak ground acceleration, and the SSE based on also CSDRS of 45 (inaudible) spectra. The(inaudible) backdrop

for SSE are (inaudible) HRHFRS is hard rock high frequency response spectra, and the peak ground acceleration is 2.2 (inaudible).

This figure shows you margin to CSDRS. Features for the CSDRS as follows. The margin to CSDRS are developed from the Reg. Guide 1.50 shifted the(inaudible) spectra, basically, and the (inaudible) value is 0.3g. Below 9 hertz, (inaudible) Reg. Guide 1.50 spectra, and at the frequence of 25 hertz the spectra values of Reg. Guide Spectra are increased by 30 percent, and for considering high frequence motion at (inaudible) site is standard from 33 hertz to 50 hertz. DC CSDRS will be discussed in detail in Chapter 3.

CHAIRMAN BALLINGER: With respect to this Figure 2.0.1 and 2, by the way, in the DCD nothing is labeled. This figure is clearer but -- and I guess we can guess what damping 2, 3, 4, 5, 7, and 10 percent are, but in the DCD there's no labels at all that tells you anything. So you probably ought to think about fixing that.

MR. KOH: I see. We will.

MR. Y. KIM: Yes, we will incorporate logistics in our DCD.

CHAIRMAN BALLINGER: Not only that, but

in terms of --

MR. KOH: You mean label the damping --

CHAIRMAN BALLINGER: All blue lines, we know -- we can guess what the damping factors --

MR. KOH: Yes, okay.

CHAIRMAN BALLINGER: Yes.

MR. KOH: Okay.

CHAIRMAN BALLINGER: Thank you.

MR. KOH: Damping labels will be added to here.

CHAIRMAN BALLINGER: Okay, thank you.

MR. KOH: This figure shows data for CSDRS. (inaudible) value is 0.3g. This figure shows you high response spectra, HRHF high response spectra. Even though the CSDRS amounts to high frequency application in the high frequency range between 9 and 50 hertz, those CSDRS are not anticipated to cover ground motion response spectra for all potential Central Eastern United States sites. Therefore, this HRHFOSO is used for covering high frequence ranging. The PGOHRF response spectra is 0.2 (inaudible), and this response spectra will be discussed in detail in Chapter 3.

This figure shows you what HRHF response spectra. (inaudible) deformation is a COL

item. The COL applicant recheck the deformation potentials and come from the (inaudible) within the design basis of the facility. For uniformity of soil, (inaudible) angle is limited to 20 degrees. (inaudible) static bearing capacity is 50,000 PSF minimal allowable, dynamic bearing capacity for normal conditions, dynamic bearing capacity is 50,000 PSF (inaudible) of soil is 1,000 feet per second, and the liquefaction potential is none for (inaudible). Minimum soil angle of internal friction is 35 degrees below the seismic (inaudible) structure. Allowable settlement of structures are as follows. Maximum allowable different set point inside the building is in principle 50 feet in any direction for the seismic (inaudible) structures. Minimal allowable different settlement between buildings which is three inch between (inaudible) basement and the EDG building, and half inch between other buildings.

MEMBER STETKAR: Let it be on the record that I have permanent nerve damage in the tips of my fingers, and sometimes my tactile sense is not necessarily what it should be. It is now on the record.

A couple of questions on this. First of

all, I -- what particular buildings are in the scope of the certified design? You mentioned the nuclear island, you mentioned the emergency diesel generator building, and the diesel fuel oil tank building. If I look at the figure, and I have to be careful because this figure is marked security-related, so I have to be a bit careful when I refer to it. Figure 1.2-1 in the DCD shows a plan of the site, and it has marked several buildings with shading that says, scope of design certification. And all of the buildings that are listed here are in the scope, as is the turbine generator building, and the compound building. But in particular, also shown in the scope are the essential service water component cooling water heat exchanger buildings on that figure. Are they in the scope of the design certification, or not? They are certainly safety-related buildings, but are they in the scope of the design certification, or not? It's a yes or no answer. Somebody here should know that. We have a lot of people.

MR. KOH: Mr. Oh, please.

MR. OH: This is Andy Oh from KHNP. And as you asked for the scope of the building as in scope for the APR1400 design certification is ESW

and CCW heat exchanger building is in scope.

MEMBER STETKAR: In scope. Okay.

MR. ROCHE-RIVERA: If I could -- good morning. My name is Robert Roche-Rivera. I'm a structural engineer, Office of New Reactors with the EIA. We actually asked the same question you have, and this will be covered during our Chapter 3 meetings. The response, the clarification we got from the applicant is that those buildings that you have indicated are -- with regards to the seismic analysis and structural design aspects they are not part of the scope of the certified design. They are not.

MEMBER STETKAR: Thank you.

MR. ROCHE-RIVERA: And in addition to that, actually, the DCD does include COL items that address that aspect and are left to the COL applicant.

MEMBER STETKAR: Thank you. So I'm now left with answers on the record that leave me equally confused as before I answered the question. So they're either in the scope or not in the scope. If they are in the scope, it strikes me that the specifications for foundation and subsurface materials should apply to them. If they are not in

the scope, it strikes me that it should be very clear that the COL applicant must be responsible for them. And I will add not only those buildings, but also the essential service water cooling tower structures because they are certainly safety-related, and I will add the component cooling water supply and return piping tunnels that go from the auxiliary building where the component cooling water pumps are out to the heat exchanger buildings because those are also certainly safety-related structures. They happen to be underground piping ducts but they're safety-related. So I think that the DCD and the Staff's review of the DCD in all areas, not just seismic, but also any specifications for subsurface materials should be consistent about what is inside the scope, and what is not inside the scope of what's being proposed here.

MR. SISK: So this is Rob Sisk, Westinghouse. And to answer that question, clearly there needs to be some reconciliation of what Robert had to say and Dr. Oh. What we will do is cover this in more detail during the Chapter 3 review. This will be specifically covered in detail, particularly on the seismology, Section

3.7 and 3.8, and we will have that clarity made at that time.

MEMBER STETKAR: But, Rob, it also, obviously, has -- you know, depending on the answers there it has trickle down or tentacles that reach into Chapter 2 here, also, and perhaps COL information items related to Chapter 2. That's -- if they're out of scope, I'm more concerned about making sure that the COL information items in Chapter 2 make it very clear that the COL applicant must meet some specific criteria for the foundations and subsurface materials for those safety-related buildings, because they are safety-related.

MR. SISK: I understand, and we will -- as Robert indicated, as well, there are some COL information items we will need to look to see if they're fully addressed, all of the concerns you've raised today.

MEMBER STETKAR: Okay. As I said, not just Chapter 3, Chapter 2.

The second question that I had relates to your Item 7B here that says, the maximum allowable differential settlement between buildings is 3-inches between the nuclear island base mat and

the emergency diesel generator building, and the diesel fuel oil tank building. Three inches doesn't sound like a lot until I start to think about piping, and cable, and vent -- I don't know if there are any ventilation penetrations, but there's certainly piping and cable penetrations that go between those buildings. How do you justify 3-inches differential settlement between those buildings?

MR. KOH: We check all the settlement of buildings, NI buildings, and other buildings. As you know, building, NI building and EDG building, in between buildings is the only electrical lines there, so that shows that 3-inch is almost --

MEMBER STETKAR: Okay. Are there only electrical lines, or are there cooling water lines for the DC generators?

MR. KOH: It is, between NI building and the EDG building, there is only electrical lines, but between and NI buildings, and other buildings, there are major steam lines and other piping. So the between NI building to EDG building amount of settlement is although with different amount to each, quite large, 3-inches, but within NI building, other buildings here, amount of

settlement is only half inch because considering the piping, venting, piping --

MEMBER STETKAR: I'm sorry. That says inside the building. That's something like --

MR. KOH: No, no, no. Here, please, here.

MEMBER STETKAR: Oh, between other adjacent buildings.

MR. KOH: Therefore, half inch between other buildings.

MEMBER STETKAR: Okay.

MR. KOH: And NI and the other buildings, (inaudible) EDG and the other buildings.

MEMBER STETKAR: So the 3-inch only applies to the EDG building.

MR. KOH: Yes.

MEMBER STETKAR: And that's because the only connections are electrical cables.

MR. KOH: Yes.

MEMBER STETKAR: Okay, thank you. That helps.

MR. KOH: Thank you.

MEMBER POWERS: And, John, we've routinely approved 3-inches. This is not an uncommon number.

MEMBER STETKAR: It's actually an uncommon number. Most of the ones we've seen is half an inch.

MEMBER POWERS: No, just last month we wrote off on 3-inches.

MEMBER STETKAR: Okay.

MR. KOH: Backfill (inaudible) properties are as follows. Density, (inaudible) ratio, and (inaudible) small strain less than 0.001 percent(inaudible) also shows (inaudible), and the damping of soil (inaudible). This figure shows you (inaudible) and damping values for the (inaudible) according to the shear strain variations as presented in the previous table. This figure is based on locally, (inaudible) Research Center report. (inaudible) research.

NRC requested many RAIs for checking APR1400 DC Chapter 2 to represent the table RAI (inaudible). This table also shows the RAIs (inaudible). This table also shows the reference (inaudible). All RAIs were addressed during the NRC Staff review and the open items were identified. (Inaudible.)

CHAIRMAN BALLINGER: I'm not sure whether this is the right point to ask this

question, but we're going in sort of order. On the DCD, Chapter 2.1 -- Section 2.1.1, talks about the exclusion area, and it talks about authority and control. And there's a statement in there that's, a highway, railroad, or waterway may traverse the exclusion area but not close enough to the facility to interfere with normal operations. What does not close enough mean? I don't know how to interpret that. I mean, a railroad track can be closer than a highway?

MR. SISK: This is Rob Sisk, Westinghouse, again. Thank you for the question. I'll need to talk with the team to give you a clearer answer.

CHAIRMAN BALLINGER: Thank you.

MR. D. LEE: This slide show the key review items for RAI Number 8012 division number 231-3. As he discussed earlier, NRC Staff note that almost all COL and ESP applicants previously reviewed identified zero percent incidence temperature greater than the corresponding site parameters for APR1400. For these RAI issues, KHNP pointed out with the temperature used by previous certified reactor which used 81 Fahrenheit degree, identified that those exceedance parameters are

same as those specified in EPRI-URD, thereby concluding that APR1400 repair temperature is reasonable and value for use. The answer for your questions will be provided for you.

MEMBER STETKAR: Thank you. And what I'd like -- I don't know whether the Staff is not going to discuss Section 2.3 today, and this is part of 2.3, but I'd like if the Staff has, or if you do, information about what fraction of the United States would not meet this criterion. We know that 21 percent will not meet 86 percent, 86 degrees. What fraction will not meet 81 degrees? And I'm not talking about land area, necessarily, because Alaska is awfully big.

MR. D. LEE: In this city, this phase is not designed yet, but the cooling tower temperature and with the others --

MEMBER STETKAR: Well, that's why I was asking about the design because that 81 degrees must be limited by some feature of the plant design. And I'd like to understand, you know, what feature of the plant design limits it to that. In other words, why cannot you -- why can you not specify a higher temperature, something like 86 to 88 degrees, which would allow you a much larger

area in Central, and Southern, and perhaps East Coast U.S.

MR. SISK: This is Rob Sisk, Westinghouse. Again, we'll need to --

MEMBER STETKAR: Yes.

MR. SISK: -- confer, as we talked about earlier, John. But I will say that, don't necessarily presume that that's a design limit as much as a calculation limit that was done at the time based on the URDs and based on the analysis of record.

MEMBER STETKAR: Yes, I understand that, but if I look at -- that one was in the ABWR certified design, and we haven't built any ABWRs except South Texas, which is kind of the ABWR modified a bit. But at South Texas they had to increase that temperature, in particular, to 88 point some odd degrees.

I think that in your slide here you also say it was U.S. EPR proposed a temperature of, I have that in my records here, actually 80 degrees as a 1 percent exceedance value. I could not find a zero percent exceedance value in the EPR DCD. It's a 1 percent exceedance value of 80 degrees, so the zero percent exceedance value would be somewhat

higher than that. How much higher, I don't know. And I'll remind everyone that the NRC has not yet certified the US EPR, so that temperature is somewhat -- could be subject to change if it's ever certified. The next lowest temperature I could find, in fact, was 86.1 -- or 86 degrees for the US APWR, and that was the one where the Staff cited 21 percent of the stations could not meet that. And then AP1000 is 86.1, the ESBWR is 88, so you could get the notion. From KHNP, in particular, I'm interested in understanding what features of the design limit that temperature. From the Staff, I'll be more interested in terms of what fraction of the United States could the plant be built given this limit.

MR. D. LEE: Next, I can make the number --- RAI number 7912, and the question number 234-1. Through this RAI, NRC Staff raised the issue as follows. Since both, the 3 and 4 -- the 3 and 4 and in the same wind friction window from the thirty point seven. As you could see, 3 and 4 is located within this same window friction. At the time the Reg. Guide 1.194, the criteria is not met, so cannot be used the reduction factor of 2. Therefore, KHNP found the fares which have same

application with the example in this RAI and update all of the analysis without credit of the reduction factor of 2.

Let me introduce summary. Chapter 2 defines site parameters of APR1400 standard plants and also identified all important site parameters. The COLA is to come from the DCD site parameters envelop site specific parameters. Table 20-1 identifies the site parameters for the APR1400 design. Thank you for your attention.

CHAIRMAN BALLINGER: Questions from Members? Thank you. I think we are now, if I even account for my dyslexia, we're actually ahead of schedule. We're scheduled for a break at 10:30. I would propose that we break from now until 20 minutes after and then pick up. Is that agreeable for everybody? Okay, let's take a break for 15 minutes.

(Whereupon, the above-entitled matter went off the record at 10:04 a.m. and resumed at 10:19 a.m.)

CHAIRMAN BALLINGER: And I believe KHNP wants to clarify one question.

MR. OH: Yes. Mr. Speaker, I'd like to clarify --

CHAIRMAN BALLINGER: Can you identify yourself please?

MR. OH: Yes. Andy Oh, KHNP Washington office. I would like to clarify my answer again for the scope of ESW and CCW. Actually, the ESW and CCW systems is in scope for the APR 1400 design.

However, the building regarding the first seismic analysis and instructional analysis is out of scope.

MEMBER STETKAR: Is out of scope. Thank you. That clarifies that and I'll keep it on the record then that I want to be sure -- I would like to have assurance that indeed in Chapter 2, which is what we are discussing today, there is -- it's adequately clear such that the combined licensed applicant understands what limitations they must meet in terms of sub-surface soil structure, foundation design and things like that for those structures -- the cooling tower structure, if they are to use a cooling tower, and for the piping tunnels that connect the complement cooling water system and the service water piping to those buildings.

MR. OH: However, steel systems design

is in scope.

MEMBER STETKAR: The system -- I understand the system in terms of flow rates and piping sizes and things like that. But I am talking about structures here.

MR. SISK: This is Rob Sisk Westinghouse, John. John, just to communicate clearly so we capture the idea correctly for Chapter 2, you're referring to the tunnels in the building. You just want to note that they are -- need to be addressed?

MEMBER STETKAR: I want to make sure that the COL applicant understands what they need to meet in terms of settlement requirements, differential settlement requirements, subsurface soil requirements for those structures because they are indeed all safe -- those -- the tunnels -- the CCW heat exchanger and ESW pump building buildings and the cooling tower buildings, if they are going to -- if they are going to use cooling towers.

MR. SISK: Okay. Let's continue. Turn on the microphone. Either that or speak more loudly.

MR. ROY: My name is Tarun Roy. I am the NRO project manager responsible for

coordinating staff review of APR 1400 Chapter 2 of the design certification application. The NRC technical staff involved with the safety review of the APR 1400 FSAR Chapter 2 RS are as follows. We have Seshagiri Tammara, then Mike Lee on the end, and Ricardo Rodriguez. We will not be presenting 2.3 today. It will be on October 4th of 2016.

During this briefing, the staff will present the Chapter 2 site characteristics with open items although there are no open items. The staff issued the 33 questions to the applicant, requesting additional information. All questions are either confirmative action or closed.

However, the last -- late last week in Chapter 3 a draft -- Chapter 3 a draft RAI response was issued from the applicant, RAI Number 8299, question 3.7.2.9. The resolution of this response may affect the DCD section 2.5.4.5 and COL item 2.5.8. RAI response will be discussed during the presentation of section 2.5. Ricardo will do that.

With that, now I turn the presentation over to the technical reviewer, Rao Tammara, for the Section 2.1 and 2.2 of the radiation protection accidental consequence branch.

MR. TAMMARA: My name is Tammara Rao.

MALE SPEAKER: Is the button pushed?

MR. TAMMARA: I am with NRO. My responsibility is to review the FSR Sections 2.1 and 2.2. 2.1 entails the geography and demography. It's the subsections covering the site, location and description, exclusion area authority and control and operation distribution.

These -- this information mainly pertains to site specific and not design specific, therefore the information is to be addressed by the COL applicant who references this design.

Therefore, COL applicant will address this information as a part of COL information 2.1 in the COL application.

Next slide, please. Section 2.2 deals with the nearby industrial transportation and the military facilities. This review also has three subsections which covers the locations and the routes nearby the site and the description of all the facilities or with the processes, follow-ups and the operations and the actions and the other section where they devaluate the potential impacts of those facilities on the safe operational plant whether there are any potential impacts. These are also, again, site specific. They are not design

specific. There are no site parameters or characteristics defined in 2.0 table by the design. Therefore, COL applicant to -- who addresses --- who references this design is to be providing this information as a part of COL item 2.2-1 and also 2.2-2 and staff will -- finds this acceptable because it is site-specific information.

MEMBER STETKAR: I have a standard question that I ask of every new plant so I have to ask it on this one to get it on the record.

In the SER you note that the following principal types of hazards will be considered with respect to each of the above areas of review. We are talking about these nearby hazards.

If they have a probability of occurrence greater than 1 times 10 to the minus seven per year and in the DCD the applicant uses that 10 to the minus seven value and said well, if we can't do a very detailed analysis we will use a ten to the minus six screening value. Revision 2 of regulatory guide 1.200 notes that it is recognized for those new reactor designs with substantially lower risk profiles, EG internal events, CDF below ten to the minus six per year, that the quantitative screening value should be adjusted according to the

corresponding baseline risk value.

So I'll ask you in that context why does the staff accept a ten to the minus seven absolute screening value for the APR 1400 design or if a -- an approximate analysis is used -- the ten to the minus six screening value?

MR. TAMMARA: In the evaluation generally we screen -- I mean, the SRP guidance is also the design basis accident is defined as the -- those events which have the probability of greater than ten to the four minus seven having a dose greater in excess of 10 CFR 100. That is the -- that is the definition of the design basis accident.

So whenever we are looking at the evaluation of the facilities, which have the potential to impact the plant, we are screening out on that basis first.

But however, if you have a detailed or more statistical data available, the subset of that definition says you can go as low or as high as ten to the four minus six on this screening. So that is the rare -- the design basis accident is defined so that is the criterion that is being used.

Therefore, when they are saying that

they will meet ten to the four minus seven, we are accepting that one. But however in the COL phase we are evaluating whether they are meeting that ten to the four minus seven. If they are not, if they are meeting the ten to the four minus six, if they are not then we have to delay the NPRA or some other mitigating measures or something else. So that -- those are the steps which are being evaluated in the review.

MEMBER STETKAR: And that's -- and I understand that and I understand why you're constrained by what's written in the SRP.

We have seen plants and we have not yet reviewed Chapter 19 of the APR1400 design certification so I have no idea what their core damage frequency is or what the basis for that is.

Recognizing that in many cases they cannot evaluate external hazards like seismic or flooding or high winds or those things until they actually have a site in the PRA.

We have reviewed new reactor designs where -- who profess that their entire core damage frequency is less than ten to the minus seven per year. So therefore screening out something at ten to the minus seven per year would be equivalent of

screening out something that's equal to everything else -- everything else that could possibly contribute to core damage or releases.

And what I am questioning is is that appropriate. I understand what the SRP says.

MR. TAMMARA: But that is present guidance available. That's how we rely on the regularity requirement.

MEMBER STETKAR: And that's why I asked the question to get it on the -- to get it on the record.

In Section 2.2.3.3 -- this is a question about a particular analysis -- it said the following principal types of hazards will be considered with respect to each of the above areas review. If they have a probability of occurrence of greater than 10 to the minus seven per year, which I just -- and two of them are missiles more energetic than the tornado missile spectra and pressure effects in excess of the design basis tornado and what I am curious about is, is it intentional for the staff to limit those missiles and the pressure effects -- the wind loving pressure effects to tornadoes because we know that in the United States there are some locations in

particular -- southeastern United States and along the Gulf Coast were in fact hurricane missiles and perhaps hurricane wind blowing are more limiting than tornado missiles. So was limitation in the SER to tornado missiles intentional?

MR. TAMMARA: I think that is covered under Section 2.3 meteorology and hydrology. But 2.1 and 2.2 are typically manmade hazards, not natural hazards. Natural external.

MEMBER STETKAR: Yes. I got it.

MR. TAMMARA: So therefore I cannot probably answer your question precisely. We might have to rely on --

MEMBER STETKAR: Well, but in 2.2, because that's your area, you're saying that if I have a manmade hazard and I don't want to be dramatic, but a local hazard that can -- manmade hazard now that can generate a missile, point it at the plant, that as long as that missile is less energetic than a tornado missile I am okay.

And I am saying that may I should be using tornado or hurricane because the hurricane missiles may be more limiting. I don't know.

I am trying to understand why you limited those manmade hazard missiles to comparing

them with tornado missiles rather than any external hazard generated missile which could be a tornado or a hurricane.

MR. TAMMARA: No, the first step is to find out that the accident probability has ten to the four minus seven it is -- if it is screened out there is no question of evaluating the --

MEMBER STETKAR: Okay. So I got ten to the minus five event that throw -- hurls a missile at the plant. So I can't screen it out. It's less than ten to the minus. It's ten to the minus five. It hurls a missile at the plant. Do I compare that missile to a tornado generated missile or to a hurricane generated missile?

MR. TAMMARA: We'd probably we'd have to evaluate the energy and see and compare. That is the -- that is the part of the analysis. We cannot throw it away but we are to -- we are to evaluate that situation and the energy of that missile. But so far that hasn't arised in our evaluations thus far. But it may. I do not know.

MEMBER STETKAR: That's right, because I don't know where they are going to put this plant and I have no idea what manmade missile hazards this plant might be exposed to, neither the

frequency nor the size nor the energy of those missiles. We don't know.

MR. TAMMARA: But that will be the -- that will be the case when the COL applicant has to address in the evaluate.

MEMBER STETKAR: We are not communicating here. I am questioning whether the comparison -- the energy comparison of those missiles that I am hurling at the plant now ten to the minus five per year so I can't screen them out on frequency, whether that energy comparison should be made against tornado missiles, which is specified in the SER or against any externally-generated missile which would include hurricane missiles.

MR. TAMMARA: Why are you are comparing that kind if you have -- if you have calculated the missile you will have the missile and its impact. We are not comparing and throwing away based upon the other missiles.

MEMBER STETKAR: This says that I can screen it out. If it's ten to the minus five I can screen it out if it's less energetic than a tornado missile.

MR. TAMMARA: If we -- if you can

demonstrate, yes.

MEMBER STETKAR: Doesn't say anything about hurricane missiles.

MR. WARD: Excuse me. I want to be clear just so we can answer the question properly. But were you saying that you think hurricane missiles may be more energetic?

MEMBER STETKAR: Oh, I know hurricane missiles may be more energetic because that's why we revised the high wind criteria.

MR. WARD: So it doesn't exceed a tornado, which you think is less energetic.

MEMBER STETKAR: That may be the rationale. But I am trying to get the staff to answer why this is specifically focused on tornado missiles.

MR. WARD: The other question I will have to check will Chapter 3 is if it's a tornado missile that was analyzed in Chapter 3 then that's what they would look to as being the limiting case an perhaps it is Chapter 3 that needs to consider that.

MEMBER STETKAR: It may be. Again, we are going chapter by chapter here.

MR. WARD: Right.

MEMBER STETKAR: And I am sensitive to this because there have been COL applicants who have had to change things because they're more limited by hurricanes in particular South Texas but and I think Turkey Point was more limited -- Turkey Point was more limited by hurricanes.

So it's an issue that's come up that the staff and us, you know, traditionally through most of the United States looks only at tornado missiles because you very quickly get past the areas that are limited by hurricanes.

There may be a rationale. There actually may be a rationale, Bill, about why tornadoes --

MR. WARD: Well, I know recently they upgraded the highest tornado wind speeds. I don't know the details but that may be the reason.

MEMBER STETKAR: Anyway, that's --

MR. WARD: We will look it up.

MEMBER STETKAR: Yes, take a look at it.

MR. TAMMARA: Thank you. That's all my presentation.

CHAIRMAN BALLINGER: So who's next?

MR. LEE: I am.

CHAIRMAN BALLINGER: So who's next? You are. Okay.

MR. LEE: So if we can go to slide ten. Hi. My name is Mike Lee. Good morning. I was given the hydraulic engineering section to review.

As you can see, this list of items that the COL applicant must address in their application is not unlike what you might find in NUREG 0800.

All these areas rely on-site-specific information or scenarios that the applicant, the COL applicant, would collect and present at the time of the -- of an application.

Upon inspection, you'll see though that geochemistry groundwater -- geochemistry is not one of these items that's come up in an earlier discussion and we'd be happy to take that question back and get back to you on that at the next time we come before you, if that --

MEMBER STETKAR: I didn't -- you know, I understand it's on the record. I didn't want to repeat it here.

MR. LEE: We will have an answer because when Chapter 3 kind of shows up there may be some open items. So we will review with the subcommittee so we'd be happy to go ahead and do

our homework on that.

So if we can go to slide 11. Thank you. All right. So as I said, the information on the previous slide is what the -- it's site specific and was provided by the COL applicant and there is guidance out there on how to do those reviews. For example, you have Reg Guide 1.206 and more recently you have JLD ISG 2012.06 which the staff is currently using to review the 50.54 evaluations from the existing fleet.

As was mentioned earlier this morning, there is a number of parameters that have already been reviewed and commented on by this or an earlier subcommittee related to groundwater levels tsunami and precipitation rates relative to rainfall. So in addition to collecting information about the site that's based on geology and meteorology and climatology and the like, the COL applicant will have to also meet these -- this design envelope. So if we can go to slide 11 or 12.

MEMBER STETKAR: I hate to do this to you. Let me -- let me just make sure it's clear because the applicant about this notion of what do they mean in the vicinity of SSCs that are

important to safety that's sort of a different philosophical question.

Let's get back to kind of a more traditional question. Are we confident that the COL applicant, if for example, the service -- central service water complement -- cooling water heat exchanger building's cooling towers, if they use them, or however they -- wherever they get their water from and those underground safety related piping tunnels.

Are we confident that the COL applicant knows that they need to meet the two feet and one feet criteria for those safety-related structures also or do they apply or do they not apply? Because everything I see is indexed to the nuclear island. Put that up on a hill.

MR. LEE: Yes. No, that's a good question and I think within the nuclear -- the power bloc there is going to be a designation of what structure systems are important to safety and also assessment.

MEMBER STETKAR: But here in particular for a COL applicant do I know that if I put the nuclear island up on top of a hill and satisfy everything in terms of flooding groundwater and

down at the base of the hill, and the hill might be pretty big, I have my service water building and tunnels that go from the stuff up on the hill to down below, do I need to meet the same flood and ground water levels for those service water buildings because they are safety related.

MR. LEE: In my judgement, the short answer is yes. I mean --

MEMBER STETKAR: In my judgement that is too. I want to make sure -- that's my own judgement. This is a subcommittee meeting. It's not an ACRS meeting.

MR. LEE: And I can tell you from our experience in the 5054 review that issue hasn't been ambiguous.

MEMBER STETKAR: Yes. Well, I just want to make sure that I couldn't necessarily find -- because I couldn't determine what was in scope and out of scope I couldn't necessarily find COL information items in Chapter 2. Not Chapter 3 but Chapter 2 that alert the COL applicant to the fact that they need to meet these things. And, you know, it might be obvious to a lot of people but a COL applicant might say well, it wasn't in the DCD, I don't need to necessarily meet it because it

wasn't in the certified design. So staff, you can't ask me about it.

MR. LEE: I think the COL applicant will be hard pressed to argue that some off -- you know, just some structured system component that was important to safety that wasn't necessarily in the power bloc physically like a service water intake structure or something like that wouldn't be subject to these criteria.

MEMBER STETKAR: I think they would be hard pressed. It's just that we are reviewing the design certification and making sure that it specifies clearly what is in the certified design and what is the responsibility of the COL applicant. Essentially no surprises because of the COL applicant.

MR. LEE: Let's hope not.

MEMBER STETKAR: Okay. Thanks.

MEMBER SKILLMAN: Mike, I'd like to build on John's earlier comment relative to the limitation that KHNP put on maximum ground water and flood.

The additional phrase for the SS or for the areas close to SSCs or the buildings close to the SSCs is an add on from the design cert from

combustion.

And so at least in my view that little phrase carries an immense burden and it would be prudent if that phrase was not there.

What is near to me or near to you or what's near to one of my colleagues might be very different than what our colleagues from KHNP think is near and I can just imagine the torture that we would go through to determine what the rate of ingress is of ground water for something that's 100 yards away or 200 yards away or 500 meters away. It just seems like it poses a question that does not need to be addressed.

MR. LEE: Enters as an ambiguity.

MEMBER SKILLMAN: It does. So I just want to reinforce John's earlier comment on that.

CHAIRMAN BALLINGER: I think that's a question that we can take back with us. Thank you.

MR. LEE: All right. I am on slide 12. All right.

So there is one COL information item which is information item 2.4.1 and it stipulates that the applicant is to provide -- the COL applicant is to provide sufficient information to verify that the hydraulic related events will not adversely affect the safety basis for the APR1400

design.

The COL applicant is to characterize the potential flooding hazards that may affect the proposed reactor site and the COL applicant decides on what combination of design features and that could be elevation of door openings, the addition of field to raise the power block raid, passive transit system berms or administrative procedures which will be used to mitigate the effects of any potential flood hazard at the proposed site.

So the items that are identified here are not unlike what we are currently seeing in our reviews of the 50.54 flood hazard evaluation. So there is nothing new here.

So in closing, if we can turn to slide 13, the DCD applicant has provided plant-specific hydrologic site parameters the staff finds as acceptable, as I mentioned earlier. Those have already been reviewed by this committee or an earlier -- or this subcommittee on an earlier.

The applicant -- the DC applicant has properly identified the site-specific information be provided as part of any future COL application and the applicant has -- the DCD applicant has satisfactorily answered all earlier RAIs and there

are no open items. So be happy to answer any questions you might have at this point. Thank you for your time.

MEMBER STETKAR: No questions?

MR. RODRIGUEZ: So hi. My name is Ricardo Rodriguez. I am a geotechnical engineer NRO DSEA. I am presenting Section 2.5. Just a little caveat. A team of people is actually working 2.5. Stephanie Devlin-Gill, David Heestel, Luissette Candelario, they are both seismologists and geothermical engineers as well. So if there are any questions I might rely on them for the response.

So slide 14 talks about essentially what we cover in our review. Range is basic geologic and seismic information, background information, surface deformation, stability of subsurface materials and foundation, and stability of slopes. Essentially what we did here is as a caveat for the next slide please -- so what we are looking at this -- at this -- for this section what we our focus wants frank parameters that are representative of a potential COL or ESP site. So the applicant had to demonstrate that the parameters as they described in the section in

Table 2.01 were representative of a reasonable number of sites that they were included as part of tier one.

There was a summary table included and also not all the focus of the RAIs where the applicant needed to provide the basis for those parameters. As you can see here, some of the parameters that we are -- we look at are that first line should actually read allowable bearing capacity not demand. Minimum shear wave velocity allowable level of seettlement as the applicant discussed this morning. No liquefaction potential for seismic category one structures. The potential for surface deformation as I said earlier this parameter is expected to be representative of a potential COL or ESP site.

Next slide. So I'll talk about the findings we have for the first three sections. We found that DC required the site specific geology considerations are addressed by COL applicants in accordance with the regulations.

Also, the DC requires COL requirements to conduct site specific PSHA and site response analysis in accordance with relevant regulations We had a number arise from this

issue but it was the result at the end. The site specific sub-surface profiles need to be compared to the profiles in the DCD to ensure that the parameters in Chapter 3.7 are relevant to COL sites.

For finding the DCD required the COL applicants show no potential for tectonic or nontectonic surface deformation. Any questions? Next slide. So the two more geotechnical engineer related sections are 2.4 and 2.5. We as a number of RAIs -- the majority of them were related to soil or rock informing the requirements. The applicability of static parameters especially related to the structure of fill, complications regarding the calculation of carrying capacity in sediment.

We wanted to make sure that the protection are adjacent to and under seismic categories was addressed and also for 2.5 we asked and RAI providing asking the static events the stability of all natural manmade slopes.

In summary, the applicant adequately addressed RAIs in many instances. They created COL -- new COL information items tasking the future COL applicant to authorize the issues at hand.

And on a related note, as we talked about earlier, we were aware that the essentials of this water building were out of scope. We had that very same question at the very beginning. We had -- we had a real issue with the SED people so we were aware of that. And so we reviewed this section with the understanding that there will be signed by the COL.

MEMBER STETKAR: That's -- yes, and I got that. I was fairly confident in that answer when I asked the question. But I wanted to make sure that the applicant was on record clearly because, as I said, the first thing that I look to -- because essentially a lot of the stuff in Chapter 2 was silent on those buildings.

Paid a lot of attention to the nuclear island in particular and the diesel generator building. And said, well, what about those buildings.

So the first thing I looked at was the figure that I referred to in Chapter 1, which to me said they are part of the certified design because they are buildings. It wasn't flow diagrams. And my bigger concern now is to make sure that it's really clear in the COL information items in both

Chapter 2 and Chapter 3 what the COL applicant needs to assure regarding those buildings and the connecting tunnels.

MR. RODRIGUEZ: Right. Right.

So I know for a fact there is -- I can't remember the name of the COL item number but there's a COL item that's state -- that the COL began has a responsibility essentially to decide those two other buildings.

MEMBER STETKAR: Well, we -- I didn't look at Chapter 3 because I ran out of gas. But, again, Chapter 3 may be just structural design -- structural and seismic design. It doesn't necessarily address subsurface soils and fill and ground water and flooding levels and all of that kind of stuff which are Chapter 2 types of things and wind blowing and all of that kind of stuff. Well, wind level is probably Chapter 3.

MR. RODRIGUEZ: Yes. So next slide. So these are the completion that RDS staff came up with. So there are new open items that are part of the Chapter 2 review.

There are 15 information items out the five -- Chapter 2.5 that adequately represent the actions needed to be taken by the potential COL

applicants and the applicant identified -- appropriately identified appropriate geologic, seismologic, and geotechnical site parameters and site specific formation to be provided as part of any future COL application.

And the staff is currently tracking confirmatory items. Also as I mentioned at the beginning, we recently were made aware last week actually there was a direct response of an RAI -- subject RAI that may have some impact into Section 2.5 and we are currently tracking the resolution of that RAI. So I have no other details than that because it's still draft. But it's a note for the committee. That is -- do you have any questions?

CHAIRMAN BALLINGER: We are remarkably ahead of schedule by an hour. And so we have, I think, two choices. We can go to lunch now and come back at noon or we can pick up and begin Chapter 5 assuming that KHNP and everybody can do that. I'd ask people's advice what do we think we should do.

MEMBER STETKAR: Oh, some of us have -- some of us have a meeting at 12:00 so --

CHAIRMAN BALLINGER: Including myself. Including myself. Okay.

MEMBER STETKAR: But you don't need to be there.

CHAIRMAN BALLINGER: So that reduces the number of options to one, which is we start. So can the KHNP people -- can you shift out? Is that going to work?

MR. SISK: What's the final plan?

CHAIRMAN BALLINGER: Oh, the plan is to push forward. Very good. Thank you.

MR. WARD: Excuse me, can we have five minutes to get Chapter 5 down from NRC here?

CHAIRMAN BALLINGER: We'll take a five minute hiatus.

(Whereupon, the above-entitled matter went off the record at 10:57 a.m. and resumed at 11:05 a.m.)

CHAIRMAN BALLINGER: Okay. It's yours.

MR. SISK: Okay, Dr. Ballinger. Thank you. Rob Sisk with Westinghouse again. We are ready to go with Chapter 5. I think given the time allotment; and hopefully have a good discussion for this section, we think we can get through 5.2 before lunch. I understand you have a noon deadline, so we will try to cover 5.2 this morning and then we'll cover the rest after lunch or at the

next available timing as it works out. So if that's acceptable to the ACRS, I'll turn it over to Minseok Kim and he will lead us through Section 5.2, or Chapter 5.2.

MR. MINSEOK KIM: Okay. Good morning. My name is Minseok Kim. I'm one of the Chapter 5 engineer. I really appreciate the opportunity to participate this meeting and present Chapter 5.

Let me start. The contents of today's presentation for Chapter 5 is on the screen. Firstly, the overview of Chapter 5. And then the reactor coolant system and also safety connected systems section by section will be presented, including the open items. Finally, I will summarize. And that's the end of the presentation.

This slide shows the list of submitted documents. Presented here for Chapter 5 itself and Tier 1, there are four technical reports. There have been 77 questions and all of the questions were responded by KHNP. There are six questions for RCP were submitted after the issue of the SCE.

Chapter 5 section overview. Section 5.1, Summary Description, describes the design features of the reactor coolant system. 5.2, Integrity of RCPB, describes the measures that

provide and maintain the integrity of the RCPB throughout the facility's design life. 5.3, Reactor Vessel, describes the reactor vessel material issues. Finally, 5.4, RCS Components and Subsistent Design, describes the reactor system components.

Okay. Here are three of today's presenters shared 5.1, Summary Description. The RCS and the components are shown in the figure. On the left of the slide the design features of APR1400 are listed. It includes 60 years design life and primary and secondary operating conditions. APR1400 is a pressurized water reactor with two closed loops. The major components are the reactor vessel, two parallel heat transfer loops each containing one steam generator and two RCPs, a pressurizer connected to the hot leg on loop No. 2. All components are located inside the containment building.

MEMBER SKILLMAN: Before you proceed to that slide --

MR. MINSEOK KIM: Okay.

MEMBER SKILLMAN: -- let me ask you to go back the slide 2, please. One more. Which document provides the accounting for the

transients, the number of heat-ups, cool-downs, LOCAs, main steam safety valve openings of 150-100 or 190-100 or 50-90-50 load swings? Where are the transients accounted for for the reactor coolant system pressure boundary? I do not see them in Tier 1 or Tier 2.

MR. MINSEOK KIM: Is that question for THE -- THEN occurrences? Is it?

MEMBER SKILLMAN: Rob, what I'm really asking for is your reactor coolant system functional specification bases.

MR. SISK: You understand? We'll take that as a note. We'll get back to you after lunch.

MEMBER SKILLMAN: Let me tell you why I'm asking. This plant does not have isolation valves. This plant can run backwards in several loops depending on what the pump configuration is. And I'm particularly interested in the reverse flow delta T versus time for if you drop a pump, four pumps, you drop one pump, your now idle pump runs backwards. You go back up on your anti-rotation device, but that loop is now flowing backwards. And if you are operating one pump in each loop, the idle loop in -- the idle pump in that loop is running backwards.

So I'm curious where that is addressed in your thermal hydraulic cycles as the basis for your ASME Section III, Class 1 analysis.

MR. SISK: I understand the question.

MEMBER SKILLMAN: Thank you.

MR. SISK: We have -- please.

MR. ROE: Good morning. My name is Tae Sun Roe working for KEPCO E&C.

That kind of information is included in the ------------- transient document. It's not listed in this list of submitted documents. But that document includes the transient including the one pump down and it contains the flow variation, temperature variation and pressure variation and number of occurrence for the transient.

And the question you said regarding the reverse flow, I think the Dr. -- Mr. Choi will answer on that question, right?

CHAIRMAN BALLINGER: You said that there was a document. Which --

MEMBER SKILLMAN: I would like -- if we may see that document, that would help me. That way I will understand the transient count, or the number of transients that are included for a 60-year design life for this reactor coolant system

pressure boundary. So if we could see that document, I would appreciate it.

MR. MINSEOK KIM: As I understand the description for the transients and occurrences are listed in Chapter 3, Section 3.9. Is that right?

MEMBER SKILLMAN: For the reactor coolant system?

MR. TAE HAN KIM: This is Tae Han Kim from KEPCO E&C. 3.391.

MR. MINSEOK KIM: 391?

MEMBER SKILLMAN: Thank you. I will look there. I would have thought they were in Chapter 5 for the reactor coolant system.

MR. SISK: So Rob Sisk, Westinghouse again. So please refer to Chapter 3. And there was a question still left on the table about the reverse flow that you were going to address? Okay. So I wanted to make sure we got that part of the question resolved. And then we're going to have a -- just a brief response on the reverse flow.

MEMBER SKILLMAN: Thank you. I'd like to hear that. Thank you.

MR. CHOI: Good morning. My name is Seognam Choi. I'm senior engineer of the Component Design and Mechanical Groups. And we have two

questions in April meetings.

MEMBER SKILLMAN: Yes.

MR. CHOI: Yes. And the first one is the momentum and the flow load changes in reactor coolant system. And second is the reverse flow through the RCS loop. It's accounting for the RCS loading analysis? Yes. And then we try to the analysis in progress and not -- I'm sorry is not finished yet, but in about one or two month we try to finish the report. And then we consider the full load, the mechanical hydraulic force because it by the equivalent and it's calculated and applied to the RCS coolant and some other. And then the hydraulic forces are due to the frictions and the momentum changes are caused by reactor coolant flows. And we tried to -- the in progress. I'm not telled about the district, but later then I will try to the answer that you are questions.

MEMBER SKILLMAN: In the future will be fine.

MR. CHOI: Yes.

MEMBER SKILLMAN: Thank you.

Rob, thank you.

MR. SISK: Okay.

MR. MINSEOK KIM: If I may resume on

the presentation, the slide on the functions of reactor coolant system. First one is energy transfer from the core to the steam generator to produce steam for the turbine generator. Secondary barrier to the list of fission products. And there are several more functions listed.

The principle parameters of the reactor coolant system can be found in Table 5.1.1-1.

MEMBER SKILLMAN: Could we back up one slide, please? Back one more.

Your last bullet, venting of steam and non-condensable gases. I read in the description that there's the vent from the pressurizer and there's the vent from the reactor vessel head.

MR. MINSEOK KIM: Yes.

MEMBER SKILLMAN: What is the sizing basis of the vent from the reactor vessel head? Is that intended to ensure that you do not have a gas bubble on the reactor vessel head after an accident?

MR. TAE HAN KIM: This is Tae Han Kim from KEPCO E&C. RCG, reactor core gas venting system including pressurizer venting and reactor header venting is based on the --

MEMBER STETKAR: Can you please speak

up a little bit so that --

MR. TAE HAN KIM: Yes, they have natural circulation analysis. NCC. Recall, NCC.

MEMBER SKILLMAN: Oh, natural circulation analysis?

MR. TAE HAN KIM: Yes.

MEMBER SKILLMAN: What consideration is given in that analysis to elimination of loops seals in the cold legs?

MR. TAE HAN KIM: I'm afraid that that is little bit different perspective of -- regarding the loop --

(Simultaneous speaking.)

MEMBER STETKAR: Will you please speak up so that we have you on the record? You have to speak into the microphone.

MR. TAE HAN KIM: Yes. Thank you. I think that is a little bit different design feature of loop seal. Venting of steam and condensable gas is venting from the head of pressurizer and head of RCC base, not in the reactor -- I mean, loop pipe.

MEMBER SKILLMAN: I understand that. I'm placing that question on the record for future action, please. Thank you.

MR. MINSEOK KIM: 5.2.1, Conformance

with Codes and Code Cases. The classification of RCPB conforms with the requirements of 10 CFR Appendix A Division 1. A quality group ABC in accordance with Reg Guide 1.26, and are designed and fabricated with ASME Section III for in-service -- service inspection to RCPB. 5.2.4 and 6.6 provide the description of the application of ASME Section XI.

For the in-service testing to RCPB application of ASME Code for Operation and Maintenance of NPP can be found in Section 3.9.6. Reg Guide 1.84, 147, 192 are used in determining the applicable ASME Code cases.

5.2.2, Overpressure Protection.

MEMBER STETKAR: Before you start on this, I have a general question, because you're -- no, you can keep on slide 8 -- because you're going to start talking about valves here. But I want to talk about configuration.

In Section 5.1.2 there's a general statement that says overpressure protection for the RCS is provided by four POSRVs connected to the top of the pressurizer. These valves discharge to the in-containment refueling water storage tank where steam is released underwater to be condensed and

cooled. Good.

If the steam discharge exceeds the capacity of the in-containment refueling water storage tank, it is vented to the containment atmosphere. I found that strange. That's unusual. So I looked at your Figure 5.1.2-3 in the DCD where I see that in each of the two lines from the POSRVs to the IRWST there are motor-operated three-way valves. That to me is a very unusual design. I don't think I've seen that.

So I have two questions. First of all, under what conditions is the IRWST not capable of handling a discharge from the POSRVs? In other words, what conditions would require me as an operator to vent directly to the containment?

That's the first question. Do you have an answer to that? Or if not, we can do it after lunch.

MR. ROE: My name is Tae Sun Roe from KEPCO E&C. During the design-basis instance we did not discharge to the containment atmosphere or the discharge from the POSRV goes to the IRWST. But in case of the beyond-design-basis such as a severe accident, then the IRWST cannot cool all discharge from the POSRV condensing the steam. So in that

case the operator open the survey valves and the -- discharge to the containment atmosphere. So is not within the design-base. It's beyond-design-basis such as a severe accident.

MEMBER STETKAR: Well, okay. I'm not a severe accident person. Dr. Rempe and the good Dr. Powers, as opposed to the bad Dr. Powers, are more severe accident.

When you say "severe accident," what in particular do you mean by that? Because if you're relieving the energy into the containment, I still have to remove the energy from the containment. So I don't understand why it makes a difference whether I'm blowing it into the containment or whether I'm blowing it into the IRWST.

MEMBER POWERS: The containment pressure goes up.

MEMBER STETKAR: Because the containment pressure would go up either way, wouldn't it?

MR. OH: This is Andy Oh, KHNP Washington Office. And during the severe accident lot of hydrogen can be generated from the melted down core, so if that hydrogen is inputted in the IRWST and IRWST is -- lot of hydrogen can be

accumulated in a small amount of the volume. So in order to prevent that event we have use three-way belt to vent it out to the containment to atmosphere to the preventing some of the high accumulation of the hydrogen inside the IRWST.

MEMBER STETKAR: Thank you. So it's basically a hydrogen vent?

MR. OH: Yes. Three belt is major role is venting for hydrogen, the preventing of the accumulation of hydrogen inside the IRWST.

MEMBER STETKAR: Okay. Thank you. I don't know whether anyone else has any questions on that, because I have a separate question.

Because it's very unusual to have motor-operated valves in the discharge flow path from the primary safety valves, I'm curious whether it is physically possible for those valves to block flow. And I don't want a licensing answer. I don't want an electrical answer. I want to understand how those valves are configured internally such that I have assurance that regardless of how they may fail internally I can still have a discharge from the POSRVs to some place.

Because if I can't, I don't have

pressure relief from my primary system. And if one of them is blocked, I don't have half of the pressure relief from my primary system. So I really want to understand how those valves are configured internally and how you have assurance that indeed they are always open to someplace.

MR. TAE HAN KIM: This is Tae Han Kim, KEPCO E&C. I think you questioning the three-way valve blocking case. Any possibility -- is that right, your concern? Any possibility of blocking the three-way valve?

MEMBER STETKAR: I'm interested in the possibility of a three-way valve being positioned such that it can block or severely restrict flow.

MR. TAE HAN KIM: We design and -- for procurement we gave it some kind of requirement not to have --

(Simultaneous speaking.)

MEMBER STETKAR: I didn't see any discussion of these valves whatsoever in the DCD, nor did I see any discussion of these valves in the SER. And that's why I'm really curious about these valves, because I haven't seen any valves like this in any other design that I've ever looked at.

MR. MINSEOK KIM: Minseok Kim speaking.

In Chapter 6.8 is there any description for three-way valve there? Please somebody check this.

MR. SISK: This is Rob Sisk, and for the sake of time we're going to take a look and get back to you after lunch.

MEMBER STETKAR: Yes, if you can.

MR. SISK: We'll continue on.

MEMBER STETKAR: I searched a few places and I did some word searches. I think I looked at Chapter 6, but I may not have done the right word searches.

MEMBER SKILLMAN: I would like to build on John Stetkar's question, please. I'm in your Tier 1 document at Figure 2.4.1-2. It's on page 2.418. And this image shows the four POSRVs relieving to the IRWST but also relief to the RDT, the reactor drain tank. And I searched on reactor drain tank and I do not find the reactor drain tank described anywhere in the document. So I'm curious, what is the RDT versus the IRWST?

Furthermore, as I read the documentation, the piping from the POSRVs to the IRWST is not seismic. And I'm curious why that piping is not seismic. Two questions.

MR. MINSEOK KIM: The drain tank is --

I think it's -- it can be found in Chapter 9. The drain tank is -- it belong to chemical and volume control system. And --

MEMBER SKILLMAN: I'll be happy to look, but I don't think it's mentioned in the SER. Now maybe in the SER for Chapter 9, but it's certainly not in the SER for Chapter 5, even though it's identified as part of your relief system off the reactor coolant system.

MR. TAE HAN KIM: This is Tae Han Kim from KEPCO E&C again. This (inaudible) line into the Tier 1 is showing the -- just a kind of a leakage indicates of line we have leakage. Then we flowing down to the RDT. This not the main purpose of discharge to POSRV.

MEMBER SKILLMAN: But it's described in Chapter 9, you say, Auxiliary Systems?

MR. TAE HAN KIM: I think so, but I have to confirm that.

MEMBER SKILLMAN: Let me leave those questions on the record, please. Thank you.

MR. OH: This is Andy Oh, KHNP again. For -- in terms of three-way valve, that classification is described in Chapter 3, and also the function is described in Chapter 19.

CHAIRMAN BALLINGER: So I'm also assuming that the issue related to severe accidents and power to these valves is discussed in one of those chapters. Is that correct?

MR. OH: Yes. Andy Oh again. I think so, yes.

MEMBER STETKAR: I'll look at Chapter 3 and Chapter 19 over lunch to see what they say about those valves, but my suspicion is given what's typically in those chapters they're not going to describe what the valves look like. I want to understand can those valves block or severely restrict flow? Is there any way that they can block or severely restrict flow?

CHAIRMAN BALLINGER: Regarding the seismic qualification I was trying to search through that, and there's a difference between Seismic Category 1 and Seismic Category 2 for that piping that's in one of the documents somewhere, but it's unclear.

MR. SISK: And just -- this is Rob Sisk, Westinghouse. Just to address Gordon's comment, if you take a look at Section 9.3.3.2.4, Component Descriptions, it does go a little bit into RDT. I don't know whether it would address

all your concerns, but there is a bit of a discussion.

MEMBER SKILLMAN: Yes, I'm there now. I can see it.

MR. SISK: Okay.

MEMBER SKILLMAN: I was anticipating, but since it's credited out of Chapter 5, it would show up somewhere in 5, but particularly because it's in Tier 1 and shown on that image, there would have been a description of it as part of the relief system or drain system. So thank you.

MR. MINSEOK KIM: Okay. Overpressure protection of the RCS and steam generator is precluded by operation of the pressurizer POSRVs, MSSVs and by the RPS. Pressure lift capacity is conservatively sized to satisfy the overpressure requirements of ASME Section III, Division 1, NB 7000. They are designed to maintain below 110 percent of design pressure during the worst case loss-of-load event with (inaudible) reactor trip. SSE suction relief valve provides sufficient pressure relief capacity to mitigate the most limiting LTOP events during low-temperature conditions.

MEMBER STETKAR: Are you going to

discuss the LTOP protection in other slides, or is this the only slide that mentions it?

MR. MINSEOK KIM: For the design feature this is the only slides --

MEMBER STETKAR: Okay.

MR. MINSEOK KIM: -- and there are open items regarding LTOP.

MEMBER STETKAR: That's okay. Then let me ask my questions about LTOP here. I understand the design, first of all. My question is, first question is do I have assurance that both SCS suction lines will be open to those relief valves at any time that LTOP protection is required? In other words, there's a relief valve in each suction line. And to have that relief valve provide protection I need both of the motor-operated valves from the hot leg to that relief valve -- they must be open, right?

(No audible response.)

MEMBER STETKAR: Say, yes, that's correct. I didn't find anything that specifically told me as a plant operator that I must have those motor-operated valves open. I can find things that are written in the draft technical specifications that say the relief valves must be operable. And

they give me an opening set point -- opening pressure set point. I can find things in the technical specifications that says I must have at least one shutdown cooling system in operation during modes 4, 5 and 6. I cannot find anything that actually tells me I need to have both of those lines open.

So if you can tell me where in some sort of guidance the COL applicant knows that he needs to write his procedures such that below the -- are the temperatures proprietary or are they not proprietary? The LTOP enable and disable temperatures, are they proprietary or not?

MR. TAE HAN KIM: This is Tae Han Kim from KEPCO E&C again. The LTOP valve is -- as you explained to us, it is told in the (inaudible) shutdown cooling system, and the shutdown cooling system is manually put in the operation when there is -- the RCS decrease down to the mode, low mode. So at that time the LTOP valve is automatically connected when we open the isolation valve to the RCS.

MEMBER STETKAR: Correct. I understand that, but if I can cool down using only one SCS train, I will only have one valve connected to the

system. Now I know one valve has adequate relieving pressure, but many of the discussions in the DCD seem to indicate that I have two relief valves available.

MR. TAE HAN KIM: Yes, we have two relief valves, but capacity is based on the -- how can I say? The -- two each one have a full capacity for our evaluation for LTOP relieving.

MEMBER STETKAR: Okay.

MR. TAE HAN KIM: In the case of we have only one shutdown cooling pump line available, we can have a full capacity of LTOP.

MEMBER STETKAR: I think what I'm trying to understand is is it possible, is it allowed for me as a COL applicant -- am I allowed to go into the LTOP regime if -- and I won't specify the temperature, but if I'm below the enable temperature am I allowed to have only one SCS cooling train allowed, or am I supposed to have both of them aligned?

MR. TAE HAN KIM: For normal shutdown cooling to the refueling operation we need two shutdown cooling operation for normally pre-down and maintain the refueling state. But we also have a -- based on the SRP, we -- one shutdown cooling

pump can also maintain into the shutdown state even though the little bit longer time is needed.

So for the case of two operation we have a full capacity of LTOP. And even though we have one line of SSCS, we have a -- what can I say, full capacity of LTOP by each.

MEMBER SKILLMAN: The limiting condition for operation is 3.4.11, Chapter 16, and it says two operable LTOP systems shall be operable as follows: Two operable. It then gives conditions if a valve is not operable. But it appears as though the answer to John's question is two.

MEMBER STETKAR: Well, I used to run a plant and I will tell you that if the operability requirement is that the pressure set point must be greater than 530 psig, I will say that valve is operable even though it's isolated from the reactor coolant system. It's still operable in the same way that a standby pump is still operable despite the fact that it's not running. Unless something tells me that I need by law to open the motor-operated valves, I'm not going to do that unless I physically need to do that somehow. And that's what I'm trying to establish, whether or not --

MEMBER SKILLMAN: There's something that --

(Simultaneous speaking.)

MEMBER STETKAR: -- there's something that tells me that whenever I go below that enable temperature I must open all four of those motor-operated valves.

MEMBER SKILLMAN: I see what you're saying. Okay. I got it.

MEMBER STETKAR: And I don't -- I can find things actually in the DCD and in the SER that lead me to believe that the intent is that I should open all four, but I want to have confidence that if indeed that's what's intended that the people who operate the plant know clearly that that's what's intended. Because they're going to write the procedures. And if they call cool down using only one train because they're component cooling water temperature is 47 degrees --

MEMBER SKILLMAN: Thirty-six degrees, yes.

MEMBER STETKAR: -- they're going to do that. They're going to cool down like a rock.

MR. ROE: Yes, Tae Sun Roe from KEPCO E&C. I'm going to add some statement regarding

your question or concern.

As you know, the Chapter 5.4.7 shutdown cooling system describes the -- such kind of case when you mention. When the LTOP alarms occurs, then the operator should open the suction line isolation valves if it's closed. So the overflow valves should be open in that case. I'm not sure that that helps.

MEMBER STETKAR: And I -- yes. And that might be it. And that alarm comes in at the LTOP enable temperature?

MR. ROE: Yes, right.

MEMBER STETKAR: Okay.

MR. ROE: If the temperature is below the enable temperature, then -- the isolation valves is not open yet, then --

MEMBER STETKAR: The operator will receive an alarm.

MR. ROE: -- operator should open the (inaudible) valves.

MEMBER STETKAR: Good. I read about those alarms, but I wasn't quite sure exactly what temperature they -- okay. That probably answers it well enough for me. Thank you.

I mean, I had a couple of other

questions if you can bear with me, Ron, on LTOP. I'll wait until we get to SCS for the other questions. I don't know if anyone else has LTOP questions.

MR. MINSEOK KIM: Continue. The figure shows the pressurizer POSRV. Main valve is here. Two spring-loaded pilot valves. And here the motor-operated pilot valves. The blue scale shows the automatic actuation for the RCS overpressure protection. Red shows the RCS rapid depressurization, bimanual actuation.

In the April meeting there was a question regarding the ASME quality requirement. POSRV can be valid committed by ASME Section III as the overpressure protection device. Regarding the capacity, according to ASME Section III POSRV has ASME capacity. Here is the example for the certification number.

MEMBER SKILLMAN: Yes, that was my question, and I appreciate the citation to the ASME Code. And I think Dr. Riccardella reinforced that, that there is an allowance for a pilot valve versus a spring valve. And that was the thrust of the question back in April. Thank you for answering that. Excuse me, John.

CHAIRMAN BALLINGER: Before we go to the next slide, you have a question?

MEMBER STETKAR: Oh, no. You said before we go -- I do before we go to the next slide, once we get back to the slide we were on.

CHAIRMAN BALLINGER: Oh, okay. Back to slide 9. A materials question: I was going through the list of materials that are used and everything. And as far as I can tell this is the only place where Alloy X750 is used, which is a super alloy. And I think that's true, but what I'd like to know is if that's true, what is the heat treatment used for that alloy, for that X750?

MR. TAE HAN KIM: Pardon me, sir. This is Tae Han Kim from KEPCO E&C. We didn't catch your question.

CHAIRMAN BALLINGER: The spring there, as far as I could read that spring material is Alloy X750, which is a high-strength nickel-based alloy. And my question is assuming that's true -- I guess confirm that it's true. And if it is, what's the heat treatment for that material? What condition is it in? Because certain heat treatments for that material are very susceptible to cracking. So I'm just curious as to whether

it's got just a standard spring temper, double aging, or what's the heat treatment for that material? Because there's a history in our industry where Alloy X750 has not reacted very well in some cases.

MR. JONGSOO KIM: Jongsoo Kim for KEPCO E&C. I think the Alloy X750 is usually in other increment. For example, a CDM. And that describe the heat treatment condition. And you can -- and let me check that section. There's another section, Section 4.

PARTICIPANT: Chapter 4.

MR. JONGSOO KIM: I'm sorry. Chapter 4.

CHAIRMAN BALLINGER: Chapter 4?

MR. JONGSOO KIM: Yes.

CHAIRMAN BALLINGER: Chapter 4. Okay.

MR. JONGSOO KIM: But that section is for the other components, but I think the same materials.

MEMBER STETKAR: Maybe same material, but maybe different heat treatment?

MR. TAE HAN KIM: Yes, sir. This is Tae Han Kim again. This is the kind of better specific material, so supplied by vendor. So we

will check it, what kind of material heat treatment is done. So we would get back to you.

CHAIRMAN BALLINGER: Thank you.

MEMBER STETKAR: Don't change. My first question on this slide is if I look at the blue box I see motor-operated valves on the inlet to the pilot and on the outlet from the pilot on this slide. If you can point to them with the -- if I look at DCD Figure 5.1.14-1, I see a motor-operated valve on the inlet and I see a manual valve on the outlet. So which is it?

CHAIRMAN BALLINGER: It's a magnification problem.

MEMBER STETKAR: Huh?

CHAIRMAN BALLINGER: The outlets are not motorized.

MEMBER STETKAR: Oh, those are manual valves on the outlet?

CHAIRMAN BALLINGER: Yes, It's just a number --

(Simultaneous speaking.)

MEMBER STETKAR: Oh, okay. I didn't -- it's just a valve?

CHAIRMAN BALLINGER: Yes.

MEMBER STETKAR: Okay. Good. Thanks.

I didn't --

MR. MINSEOK KIM: I think this one is motor-operated.

MEMBER STETKAR: Is motor-operated and the upper one is manual. Okay. Good.

My question about those was if I'm an operator, if there are four combinations of two valves being closed that will prevent the POSRV from operating in either the pilot-operated or the spring-loaded mode, how do I as an operator know that those four valves are open?

MR. YUNE: I'm Mr. Yune from KEPCO E&C. Let me explain the protocol with spring-loaded pilot valve, what are the variable. What are variable used in three quantity? If spring-loaded pilot valve open but pressure decrease, that should be shut down. But if not fail, then maintain open, the operator open -- close the motor-operated valve.

The other manual valve, we can check every (inaudible) the spring-loaded pilot valve set point. Then we close the manual valve. If not open, then manual open. So we do not want the main valve open. So that case we close the manual valve.Then operator can check the motor

(inaudible) valves. In MCR there's two pairs position indicated in the MCR and motor operated power removed during power operation.

MEMBER STETKAR: Okay. Power is removed from the motor-operated valves during --

MR. YUNE: Yes, during power operation.

MEMBER STETKAR: Well, if I'm an operator and during power operation my valve sticks open, then I have to go re-energize the motor to close the valve?

MR. YUNE: Yes, during -- infrequently fail close and power and close the motor valve.

MEMBER STETKAR: That's interesting. So I get to blow-down my system for a while while I go re-energize power.

Did I hear you -- do I have valve position indication in the main control room on both the motor-operated valve and the manual valve?

MR. YUNE: Yes.

MEMBER STETKAR: And the manual valve also?

MR. YUNE: Yes.

MEMBER STETKAR: Thank you.

MR. MINSEOK KIM: Before going further I'm afraid that the slide is not the final. Please

give time to change the slide. We have some modification on the slide, but we did not --

MEMBER REMPE: Mr. Chairman, why don't we use this as an opportunity for a break?

CHAIRMAN BALLINGER: Yes, I didn't understand the question. You say the next slide, which is slide 10 -- whoops, that's not the slide that I got here. Okay.

MR. MINSEOK KIM: Yes.

CHAIRMAN BALLINGER: I understand.

MEMBER SKILLMAN: I believe the gentleman said that there needs to be a correction to the --

CHAIRMAN BALLINGER: Yes.

MEMBER SKILLMAN: -- image on this slide.

MR. SISK: We need to caucus a little bit on this particular question and topic and get back. And I'm looking at time and the schedule. We could do that right after lunch, but I think we'll need to have a time to talk.

CHAIRMAN BALLINGER: In that case, I think it's a good time to adjourn until --

MEMBER SKILLMAN: Recess.

CHAIRMAN BALLINGER: -- recess, excuse

me, until 1:00.

(Whereupon, the above-entitled matter went off the record at 11:52 a.m. and resumed at 1:01 p.m.)

CHAIRMAN BALLINGER: Okay. We're back in session.

MEMBER POWERS: Is that like back in the saddle?

CHAIRMAN BALLINGER: Yes.

MEMBER POWERS: Back in the saddle again.

CHAIRMAN BALLINGER: You better keep going before this gets out of control here. More out of control.

(Laughter.)

CHAIRMAN BALLINGER: We're losing something in the translation.

Okay. Let's continue.

MR. MINSEOK KIM: For this section there are three open items in SER. Question 5.2.2 there were four sub-items, among them sub-item No. B and C regarding the LTOP. It was requested to provide the description of the LTOP analysis and provide (inaudible) result including the maximum (inaudible). KHNP responded on this date, but the

follow-up RAI was issued. The number was -- question number was 5.2.2-7. Provide the description about LTOP analysis methodology in parameters and solutions. KHNP responded that describe description about detailed methodology in parameters and solutions. And why these include (inaudible) solution are suitability conservative.

CHAIRMAN BALLINGER: We have a question that's from a member that's not here, Dr. Corradini on this topic. And he was -- I have his written comment. It says regarding the over-pressure protection relief during the LTOP transient. This on the open item. With regard to the energy input assumptions the staff issued this RAI, and the question, to address whether the energy inputs were conservative and how the RCS conditions for this transient were calculated.

He said please ask the staff or the applicant to explain what is uncertain about the energy input that is missing in the application. That's a question from him. Apparently there was some writing about the uncertainty in the relief valve setting and the energy input.

MR. SISK: This is Rob Sisk, Westinghouse. I think we -- Tae Han?

MR. TAE HAN KIM: This is Tae Han Kim from KEPCO E&C. The LTOP valve were basically evaluated for worst case, this addition and energy addition, so we calculated addition from the (inaudible) different temperature from primary to secondary side. But I'm not fully understand what's the uncertainty question.

CHAIRMAN BALLINGER: I think Member Skillman would know this better, but during that transient you're very close to solid. And so a very small change in input can make a very large change in pressure. And so the question was how is the uncertainty in that addition calculated?

MEMBER STETKAR: Ron, why don't we wait until the staff comes up?

CHAIRMAN BALLINGER: Okay.

MEMBER STETKAR: Because I have a question --

CHAIRMAN BALLINGER: You had a question about that, too?

MEMBER STETKAR: -- about that also.

CHAIRMAN BALLINGER: Okay. All right.

MEMBER STETKAR: And I wanted to --

(Simultaneous speaking.)

CHAIRMAN BALLINGER: All right. We'll

wait for the staff.

MEMBER STETKAR: I think it's better for the staff.

CHAIRMAN BALLINGER: Okay.

MR. MINSEOK KIM: Go on? The third item was to address the difference between the pre-service testing in DCD Section 5.2 and Chapter 14 test for pressurizer POSRVs. KHNP revised test plans in Section 14.2 to describe the complete pre-service testing for pressurizer POSRVs.

Okay. I'll turn over to Mr. --

MEMBER STETKAR: Let me ask you one question about the POSRVs. Are the APR1400 POSRVs rated for water relief? In other words, if they open under water relief, will they reseat?

MR. MINSEOK KIM: Both water and steam?

MEMBER STETKAR: Sub-cooled. I'm talking about sub-cooled water relief, so that if I fill the pressurizer full of water during a transient will they reseat or will they stick open? When I say "stick open," I don't necessarily mean stick fully open. I mean chatter or --

MR. TAE HAN KIM: Tae Han Kim again. With POSRV when we gave our procurement spec, we are required to qualify for both steam and hot

water.

MEMBER STETKAR: You will require steam and water?

MR. TAE HAN KIM: Yes.

MEMBER STETKAR: Thank you. Is that -- while I have you here, is that -- does that also apply for the main steam safety valves? Are they qualified for water relief or only steam?

(No audible response.)

MEMBER STETKAR: I see shaking of heads.

MR. TAE HAN KIM: I think this is different.

MEMBER STETKAR: So are main steam safety valves only steam relief?

MR. TAE HAN KIM: Yes.

MEMBER STETKAR: Thank you. That helps me later on when I review PRA things. It doesn't have to do anything with what we're talking about today. Thank you.

MR. MINSEOK KIM: Can you move onto third (inaudible)?

MR. JONGSOO KIM: Good afternoon, ladies and gentlemen. My name is Jongsoo Kim, KHNP.

This is Section 5.2.3, Reactor Coolant Pressure Boundary Materials. All the reactor -- RCPB material specification and selections complied with ASME Section III NB requirements and Reg Guide 1.84.

Low-alloy steel and carbon-steel. RCS main components fabricates the low-alloy steel or carbon-steel or austenitic stainless steel cladding. Cladding is in accordance with Reg Guide 1.43 and fracture toughness properties also meet 10 CFR 50 requirements. And RT_{NDT} is 10 degree Fahrenheit maximum for RCPB and minus 10 degree maximum for (inaudible) belt line materials. To reduce the neutron embrittlement effects of (inaudible) materials the chemical elements are controlled. The major structural materials are SA-508 Grade 3 Class 1, or 508 Grade 1A.

MEMBER POWERS: May I ask what steps you will take to ensure you don't run into the kinds of problems they had with the Doral pressure vessel?

MR. TAE HAN KIM: I'm sorry. I --

MEMBER POWERS: On the Doral pressure vessel with hydrogen blisters in it.

MR. MICHAEL LEE: I'm sorry, Dana. Can

you ask the question --

MEMBER POWERS: Well, what steps will you take to assure that the forgings don't suffer from the same problems of the pressure vessel at the Doral Plant in Belgium? It seems we have to do a little more than just specify the chemical composition.

MR. JONGSOO KIM: Well, as far as I know, the (inaudible) pressure (inaudible) in Korea made from the ring forging and with the requirements of ASME III requirements and with additional and examination. Yes, which is only examination with more, yes, method.

MEMBER POWERS: Well, of course that's exactly what the people at Doral did and subsequently found out with better ultrasonics that they had these quarter-sized flaws throughout their vessel. And one particular one of the rings, they go to make up the overall forging. And so it appears they're just saying good forging practices is not good enough now.

MR. SISK: Yes, Rob Sisk, Westinghouse. I will say that; as you know I'll make the generic statement because I think it's important, we do the -- monitor the operating experiences that we're

seeing across the fleet and the Doral experience and how that applies to experience in Korea, but now we're looking at a certified design, how we're assuring that that's not into a -- something that is --

MEMBER POWERS: Well, as far as we know --

MR. SISK: -- acceptable.

MEMBER POWERS: -- it seems to be a peculiarity of the Rotterdam forging practices, but it's not the only one. We had a problem with a French steam generator. Same thing. And what it says -- I mean, this is all hydrogen dissolution into the steel and an insufficient annealing time somehow. And so what do you do to make sure you don't get that problem?

MR. SISK: We're going to have to look into that in more detail.

MEMBER POWERS: I mean, I think my point is it's not evident to me that a chemical control is sufficient anymore for specifying forgings with low-alloy steels.

CHAIRMAN BALLINGER: To make it clear, it's possible to meet the toughness specifications, impact specifications and have this problem.

MR. MICHAEL LEE: I think we understand the comment and we will be looking to make sure that is addressed in our operating experience evaluations to make sure that if there is something that we can learn from that to enhance APR1400 --

(Simultaneous speaking.)

MEMBER POWERS: Yes, well, I mean, I think it's combatable because this problem of hydrogen was recognized back in the '50s and protocols were arranged to -- on the annealing to get rid of it, but manifestly they didn't work all the time. And I don't know whether it's just a failing at adherence or it's -- something more needs to be done, obviously.

MEMBER RICCARDELLA: I would suggest that a big part of that problem was the change in the ultrasonic techniques that we used way back in those days when those were built versus the ultrasonic techniques now. And I'm assuming that you guys would be using the state-of-the-art ultrasonic inspection --

(Simultaneous speaking.)

MEMBER POWERS: Well, I wouldn't assume anything.

MEMBER RICCARDELLA: No.

MR. MICHAEL LEE: We're doing it. I think it's again something we need to be tracking, and appreciate the comment.

MEMBER POWERS: Yes, somebody's going to ask you if I don't.

MR. JONGSOO KIM: Austenitic stainless steel. Austenitic stainless steel is controlled not to use to -- of sensitized stainless steel. So solution annealed condition and just carbon contents are applied. Cold worked and cobalt contents also controlled.

Nickel-chromium-iron alloys. Only Alloy 690 is used for nickel-based alloy. In addition to ASME Code requirements we incorporate the state-of-art industry requirements of EPRI document.

CHAIRMAN BALLINGER: I have a question. Back up to austenitic stainless steel. It just dawned on me -- yes, less than 0.065 percent or L-grade. Those are two different things. The specification on L-grade is less than 0.04. So L-grade would be 0.04 or less carbon. 0.065 is not L-grade. It's much higher. And so the welding procedures for something that uses 0.065 for the carbon filler metal would be different, way

different than using L-grade with respect to sensitization. So those two numbers to me are not -- I don't know. They don't make any sense together. Am I making myself clear?

(No audible response.)

CHAIRMAN BALLINGER: So reduce carbon content of base metals. Okay. To me that says maybe L-grade. I don't know. And weld materials. So less than 0.065 is standard stainless steel grade because that's from 0.04 to -- 0.08 is the maximum number down to 0.04, which is the L-grade. And then it says or L-grade, which would be 0.04 or less.

MR. JONGSOO KIM: Yes, we use L-grade for welding materials, austenitic stainless steel welding material. However, for the base material we applied maximum carbon content of 0.065 percent. That's not L-grade. But in some components I will talk later, but in this slide the surge lines were safe and we used L-grade. And for the surge line we use (inaudible) stabilized stainless steel. And we -- for the -- yes.

CHAIRMAN BALLINGER: Okay. And again, for weld materials L-grade -- since you're using a filler metal that's a duplex, I'm not sure the L-

grade means much. In other words, if you have a ferrite number that has to be within -- above five, or I forget. Pete would know the number better than I. I think it's greater than five or seven for a ferrite number of something.

PARTICIPANT: I'm not a metallurgist.

CHAIRMAN BALLINGER: Okay. Take my word for it. It's above five, right?

MR. JONGSOO KIM: Yes.

CHAIRMAN BALLINGER: So you're using a duplex material, so L -- there's no L-grade, there's no 308L, I don't think. So it's just confusing. It's just -- I understand a little bit better now, but low -- reduced carbon would be -- 0.065 would be reduced carbon from the standard grade.

MR. JONGSOO KIM: Yes.

CHAIRMAN BALLINGER: But it wouldn't be L-grade.

MEMBER RICCARDELLA: I think the operative phrase here is the first bullet under nickel-chrome-iron, "control on chemistry and mechanical properties consistent with state-of-the-art industrial requirements," or "state-of-the-art industry -- nuclear industry requirements." And I

think that would apply to all your materials.

CHAIRMAN BALLINGER: Well, but that's for -- okay.

MEMBER RICCARDELLA: Yes, but I mean, I think that would apply -- I mean --

CHAIRMAN BALLINGER: In fact nowadays it's very difficult to actually buy 300-series stainless steel that's not already L. Okay. Thank you.

MR. JONGSOO KIM: Yes. Okay. Castings. And identify the contents is limited to prevent similar aging. For examples, 14 percent maximum for CF8M. Comparability with reactor coolant. RCS water chemistry of APR1400 complied with up-to-date EPRI primary water chemistry guidelines. And the impurity control of chloride and fluoride in non-metallic insulation complied with Reg Guide 1.36.

Fabrication and processing of ferritic materials.

MEMBER STETKAR: Can I ask you; I read that, where in the APR1400 inside the containment do you use non-metallic insulation?

MR. JONGSOO KIM: The insulation is used at the outside of the component, not inside --

MEMBER STETKAR: Inside the reactor containment you have pumps and pipes and valves and vessels.

MR. JONGSOO KIM: Yes.

MEMBER STETKAR: On the exterior of many of the pumps and pipes and valves and vessels you have insulation. On the exterior of some of that you apparently have reflective metal insulation. I know what that is. And apparently some places you have non-metallic insulation. I'm asking you where inside the containment you use non-metallic insulation. Because this seems to tell me that you use non-metallic insulation in there. Otherwise, why does it say it complies with Regulatory Guide? So I'm just curious where you use it.

MR. MINSEOK KIM: The reactor -- Minseok Kim speaking. The reactor vessel insulation uses reflective metal insulation.

MEMBER STETKAR: Good.

MR. MINSEOK KIM: But the -- for the other components say inside the containment there -- they can use non-metallic insulation.

MEMBER STETKAR: That includes the reactor coolant loops?

MR. TAE HAN KIM: Sorry, sir. This is Tae Han Kim from KEPCO E&C.

MEMBER STETKAR: Yes.

MR. TAE HAN KIM: We applied Korean containment, so we tried the metallic insulation inside the containment. But I will check again and get back to you.

MEMBER STETKAR: I didn't catch the last. What did you say at the end there?

MR. TAE HAN KIM: We use the metallic insulation --

MEMBER STETKAR: Yes.

MR. TAE HAN KIM: -- inside the containment.

MEMBER STETKAR: And is that for everything? Does that include main steam lines and main feedwater lines and --

MR. TAE HAN KIM: Yes.

MEMBER STETKAR: -- all of the reactor coolant system?

MR. TAE HAN KIM: Yes, inside the containment.

MEMBER STETKAR: Inside the containment?

MR. TAE HAN KIM: Yes.

MEMBER STETKAR: Okay. That's what I was trying to establish. Thank you very much.

MR. JONGSOO KIM: So fabrication and processing of metallic materials. Ferritic materials are controlled in fabrication and process with ASME Code Rules and Regulatory Guides as applicable. ASME Section III and IX are used for qualification of welding procedure and personnel. Pre-heat control for Reg Guide 1.50 and cladding procedure qualification for Reg Guide 1.43 and welder qualification or limited (inaudible) for Reg Guide 1.71 are also applied.

And fabrication and processing of austenitic stainless steel. Austenitic stainless steel also controlled in fabrication and processes in order to avoid cracking or sensitizing. To avoid stress corrosion cracking austenitic stainless steel is supplied in solution and (inaudible) condition and (inaudible) through a specific temperature range. Cold working is prohibited and on non-sensitization is conformed by ASTM A262 Practice A or E corrosion test.

Welding of austenitic stainless steel we comply with Reg Guide 1.44.

CHAIRMAN BALLINGER: Go back. A262

Practice A or E, these are immersion corrosion tests. There's a whole list of them. I can see using 262 for qualifying material used as received, but I don't understand how you use 262 for an as-welded structure. So I'd be curious to know a little bit more information about how you use 262 for something that's welded already --

MR. JONGSOO KIM: Yes.

CHAIRMAN BALLINGER: -- to verify the sensitization --

MR. JONGSOO KIM: Yes.

CHAIRMAN BALLINGER: -- non-sensitization things. So that's a question that I have.

MR. JONGSOO KIM: Yes, I will explain using this the next parameter, using next parameter. And welding of austenitic stainless steel will comply with Reg Guide 1.44. And delta ferrite contains in welding materials, welding heat input and interpass temperature, and carbon content are controlled for ensuring integrity of weld heat affected zone and (inaudible).

And finally, ASTM A262 Practice A or E is used for ensuring non-sensitization. That means

the welding material is ensured by welding qualification. During the welding qualification process we complied -- we tested the ASTM A262 practice. So we qualify that -- that the welding material used in those welding procedure would not be sensitized.

CHAIRMAN BALLINGER: Okay. But I'm cautioning -- you're limiting your carbon to 0.065 or lower. Between 0.065 and 0.4 the response to the same heat input to an A262 test will be different, quite a bit different. So if you have piping that varies in carbon content within your specification, the response to the heat input that -- above which you get sensitization is going to be different. So it's -- I'm curious as when we talk about quality control how that's going to be assured.

MR. JONGSOO KIM: Yes, when we test the (inaudible), for example, welding material, we do sensitizing heat treatment. That means it's a little bit different, the low-alloy -- low-carbon stainless steel between the -- with the other -- the standard austenitic stainless steel. So we first sensitized (inaudible) welding and then we test the Practice A or E test.

MEMBER STETKAR: So is this a test program prior to doing the actual welding? Am I missing something?

MEMBER RICCARDELLA: I think he says it's part of the welding procedure qualification.

CHAIRMAN BALLINGER: Okay. So they do some --

MEMBER RICCARDELLA: Yes, the procedure qualification, you run the test and you cover your -- if the tests cover the range of possible carbon contents in the materials, then those procedures --

CHAIRMAN BALLINGER: Yes. Yes. Okay.

MEMBER RICCARDELLA: -- will work in the field.

CHAIRMAN BALLINGER: Thank you.

MR. JONGSOO KIM: Contamination control is applied during the fabrication process. NQA-1 requirements are applied for cleanliness and clean procedures. And cleaning water chemistry is also controlled.

MEMBER POWERS: I'm not sure what it means when you say "maintain cleanliness according to NQA-1." NQA-1, as far as I know, doesn't say anything about cleanliness.

MR. JONGSOO KIM: We (inaudible) by the

classification of the (inaudible) and there is a requirement for the -- the (inaudible) requirements applicable to the classification of cleanliness. Yes.

MEMBER POWERS: So what is the cleanliness requirement? I mean, you're using NQA-1 to make sure that you're staying within your specification, but I still don't know what the cleanliness specification is.

MR. JONGSOO KIM: Yes, the cleanliness specification is Class 1 (inaudible) 3, and those requirements are in NQA-1, Subsection 3.2. There should be -- there is applicable requirements how to classify the cleanliness. And there is also various requirements to that applicable requirement of cleanliness.

MEMBER POWERS: Okay. So you specify Cleanliness Category 1 and then you use NQA-1 processes to make sure that you're in compliance with that specification is what you're saying?

MR. JONGSOO KIM: Yes.

MEMBER POWERS: Thank you.

MR. JONGSOO KIM: And there is one open item in this section regarding DCD classification. NRC concerns the integrity of welding material --

fabricated using control parameters specified in DCD. The welding procedure specification of austenitic stainless steel qualified with ASME Section III and IX using those parameters specified in the DCD such as maximum heat input, interpass temperature and reduced carbon contents. In the weld qualification process ASTM 262 Practice A or E test is performed to issue the non-sensitization of welding material and weld heat-affected-zone.

Since the product welding is to be performed under a weld input, interpass temperature and carbon content is less than the value allowed by the qualified WPS, the weld metal and weld heat-affected-zone will not be sensitized severely. And we submitted RA response to NRC to resolve this concern. The Section 5.2.4 describes the in-service inspection and testing. ASME Section XI and 10 CFR 50.55(a) are applied for pre-service inspection and in-service inspection of APR1400.

ISI program of 10-year interval will be implemented. We prepare the PSI and ISI program and submit through licensing body prior to implementation of the code required examination. And code case is complied with Reg Guide 1.147.

Inspectability. The components of APR1400 will be designed to allow sufficient spaces for examination with ASME Code Section XI IWA-1500.

And qualification.

MEMBER STETKAR: I have a question about inspectability. In two places in the DCD you indicate that the bottom head of the reactor vessel is examined manually by removing insulation and that you have an access tunnel down where it comes in so that people can get down there to do those inspections. Is it actually feasible for people to go down there and perform those inspections manually?

But I recall when I was at Zion; this is many, many, many years ago, that the dose rates were typically so high down there that people essentially could not go there. So I was curious of all of your inspection descriptions that one raised a question with me in terms of how those inspections would actually be performed. Because everything else is underwater.

MR. JONGSOO KIM: We can go there.

MEMBER STETKAR: You can go there?

MR. JONGSOO KIM: Yes, we do.

MEMBER STETKAR: And you do that in

Korea? You actually go under and remove insulation and inspect the exterior of the bottom heads?

MR. JONGSOO KIM: Yes, we design the insulation with removable insulation.

MEMBER STETKAR: Yes, I understand it can be removed, but the dose rates are low enough that people can actually do that?

MR. JONGSOO KIM: Yes.

MEMBER STETKAR: Okay. Good. Okay. That's --

MR. JONGSOO KIM: And qualification. The qualifications of examination personnel, we comply with ASME Section XI, double A, 2300 and Appendix 7. Performance demonstration we -- with comply with ASME Section XI Appendix VIII, respectively. And the reactor pressure vessel head will be examined in accordance with Code Case N-729-1, modified with 10 CFR 50.55(a). And Boric Acid Corrosion Program will be also implemented in APR1400.

And now I will turn over to --

MR. MINSEOK KIM: 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection. The RCPB leakage detection systems are designed in accordance with Regulatory Guide 1.29 and conforms

to 1.45. The sensitivity for unidentified leakage can be discussed like this: 0.5 gpm can be detected less than one hour. And the methods are inventory, sump level and flow, containment air particulate radioactivity monitoring, and an acoustic leak monitoring system.

The definition of identified leakage is accordance with the Reg Guide 1.45 captured/flow-metered/conducted to collection, or collection system, or leakage into the containment atmosphere from a known source and the leakage instrumentation in the -- provided in the main control room.

No questions?

MR. JONGSOO KIM: Yes, Jongsoo Kim again. Reactor vessel materials. This slide describes the reactor vessel materials. This figures show the reactor vessel of APR1400 schematically. And it will be fabricated in accordance with ASME Section III basically, and Section II for materials and Section V for NDE and Section IX for welding will be applied. In various records will also be met during the verifications.

And the reactor vessel consists of low-allow steel or SA-508 Grade 3 Class 1 with austenitic stainless steel cladding. And fraction

mechanics -- provisions of minus 10 degree Fahrenheit or 10 degrees Fahrenheit of RT_{NDT} values are implemented to certify ASME Code and 10 CFR 50 Appendix G requirements. And CEDM nozzles and ICI nozzles are main components of Alloy 690 materials in reactor vessels. Reactor vessel material program are provided with ASTME 185 and 10 CFR 50 Appendix H.

MR. CHOI: Seognam Choi speaking. Section 5.3.2.1, Pressure-Temperature Limits. The purpose of the making pressure-temperature limits and of operating the plant following it is to prevent the reactor vessel pressure boundary materials from the non-ductile fracture during heat-up and cool-down transients. This is based on the requirement of Appendix G of 10 CFR 50 and Appendix G of ASME Section XI. This P-T limit curves are conservatively based on the end of life of 60 years and the critical limiting sections such as reactor vessel beltline, reactor vessel flange and inlet, outlet nozzles.

This slide shows the typical P-T limit curves for the end of life of 60 years. This curve we made provide an acceptable range of operating temperatures and pressure for heat-up, cool-down,

low-temperature overpressurizations and core criticals and in-service leak and hydrostatic test conditions. This these P-T limits curves are adjusted periodically if necessary to stay within the allowable stress limit during operation. The detail informations and explanation of APR1400 is reported to NRC. And these P-T limit curves is also combined license items. It can be developed based on the plant-specific data.

Pressurizer thermal shocks. Pressurizer thermal shock events happened to the pressurizer water reactor. That is the overcooling of the reactor vessel will followed by the immediately pressurization. RT_{PTS} for evaluating beltline values at the end of life are evaluated and satisfy the screening criteria of 10 CFR 50.61.

On this slide each RT_{PTS} values are well below the screening criteria. Also this detail information go to NRC. And then PTS is also core items. It can be developed later based on the plant-specific data.

Next. Upper-self energy. Upper-shelf energy is also another requirement to have more safety margin against non-ductile fracture by the irradiation. And this is also regulated by the

Appendix G of 10 CFR 50 and (BTP) 5-3 of NRC Standard Review Plans. And the change of the USE can be predicated by the NRC Reg Guide 1.99. With the initial limiting values of 75 foot-pounds the estimated value of end-of-life USE for APR1400 is to be about 51 foot-pounds, which is greater than the limiting end-of-life values. It can be developed later based on the plant-specific data.

MR. J. KIM: The Section 5.3.3 reactor vessel integrity. This slide then with the vessel of APR1400.

Then into which of this vessel is ensured by using techniques, proven techniques and well characterized materials by planned experiences.

The structure into which the reactor vessel is ensured by ASME Section 3 MB, material certification vessels examination, the handling of the reactor vessel comply with ASME codes and 10 CFR 50.

And, after initiation of power operation, the reactor vessel into which it will be issued by ISI program and the reactor vessel surveillance program respectively.

MEMBER STETKAR: I have a question, you

didn't cover it here, but your Table 5.3-7 specifies a schedule for removal of the surveillance capsules.

And, I was curious why the surveillance capsules are not removed from symmetric locations around the reactor vessel? In particular, why do you remove capsules at locations 217 degrees and 224 degrees ANC rather than in the four removals, have them symmetrically located around the vessel?

So, that must have been intentional, so, why do you do that?

MR. J. KIM: The location is determined high level of neutron fluence and those four -- those locations are the same -- but the same 40 degrees -- 45 degrees, it is the same, 45 degrees is the same. So --

MEMBER STETKAR: I'm sorry, but they're not. 217 degrees and 224 degrees, to me, are 7 degrees apart.

MR. J. KIM: Yes, they are symmetrical.

MEMBER STETKAR: If you look at Table 5.3-7, just look at Table 5.3-7. You are not removing the four capsules from four symmetric locations around the vessel. Just look at the table and you see 217 degrees and 224 degrees which

are right next to one another.

So, my question is, if that's intentional, why are you doing that? Your two spares are not located 180 degrees apart around the vessel as I would expect them to be. They're located about a 100 degree apart.

So, my question is, if that is intention? It that is what you are specifying for the certified design, why, as a combined license applicant must I follow this schedule? There must be an intention. So, why is it?

MR. J. KIM: Yes, this --

MEMBER STETKAR: If it's an error, then it's an error. But, if it's intentional, I want to understand why it's intentional?

MR. J. KIM: Yes, this is no error and this --

MEMBER STETKAR: It is not? Okay. Why, then, do you have this particular removal schedule?

MR. J. KIM: This location is, first is, highest location is first.

MEMBER STETKAR: Let me try this again.

Do you expect the highest fluence to be in the northeast corner of the plan view of the

reactor vessel shown in Figure 5.3-5? And, if so, why is it there? As opposed to anywhere else symmetrically around the reactor vessel?

MR. J. KIM: Basically, at this location is symmetrical. We -- from the vertical access, this is the 37 and 40, yes, 44 is -- those locations are all symmetrical from here from here.

So, from (inaudible) from 35, 44 and 35, 44. And, this is minus 35 and 44.

MEMBER STETKAR: I'm not understanding you. If I look at Figure 5.3-7, I'm sorry, Figure 5.3.-5, Figure 5.3-5, on the PDF files, it's page 149 on the PDF file or in the DCD, it's page 5.3-54.

If I look at that picture, I see that you are removing two of the four capsules at 217 and 224 degrees in the, what I would call, the northeast corner of that picture.

And, I'm curious about why two of the four are removed from there rather than removing them symmetrically around the vessel of the four?

MEMBER REMPE: But, the table has like six months, 18 months and so they're going --

MEMBER STETKAR: The table says I remove four. I remove them from 217, 37, 224 and

323.

MR. J. KIM: How many places do you have the highest fluence points.

MEMBER STETKAR: And, if you expect the highest fluence to be in that northeast corner of the vessel, why do you expect it be?

I can understand if you expect the highest fluence to be there that you would selectively sample from that location. Why do you expect the highest fluence to be in that 217 to 224 degree area?

MR. J. KIM: From our -- this -- my understanding is that basically those areas are the highest point and specifically the sequence and I will check and the --

MEMBER STETKAR: Okay, I'd appreciate it if you'd check because I don't know what there is about the core that would indicate why I would selectively have two samples from that location out of the four over the whole life of the plant.

MR. J. KIM: Yes.

MEMBER STETKAR: In other words, why I would not symmetric -- over the life of the plant, why would not select samples from each of the four quadrants?

MR. J. KIM: Yes, yes, I will.

MEMBER STETKAR: Okay.

MR. J. KIM: Thank you.

MEMBER STETKAR: I'm not a materials guy. I just would have expected that there would be, over the life, you would have gone around the vessel.

MR. SISK: Rob Sisk. What --

MEMBER STETKAR: Unless there's something unique that would give me an indication that I would expect to have higher fluence in that particular area.

MR. SISK: So, just for clarification, your expectation is that there would be samples around or --

MEMBER STETKAR: My expectation would be that there were samples that I would sample from, I don't care about the sequence at the moment, but 217, 323, 37 and 143, that would give me samples in all four quadrants.

MR. SISK: Or are you looking for where you get the highest radiation and, therefore, you'll get the most conservative answer coming out in sequence over time.

MEMBER STETKAR: And, if there's a

reason why you expect the highest fluence or the highest --

MR. SISK: So, that's what you're looking for?

MEMBER STETKAR: -- degradation at -- because, see, you're sampling from two that are very close to one another.

MR. SISK: Yes, two here and two here at different times.

MEMBER STETKAR: No, not just -- look at the picture and look at the table. You can't pull them up at the same time.

The sequence specifies that I remove capsules, the sequence, at 217, 37, 224 and 323.

MR. SISK: That's correct.

MEMBER STETKAR: So, remover two capsules that are located very close to one another --

MR. SISK: and, close in time.

MEMBER STETKAR: -- and, well, I mean not close in time, but --

MR. SISK: But, I mean --

MEMBER STETKAR: You, it's number one is at 227, number two -- I'm sorry, number one is at 217, number two is at 37, number three is at

224, which is close to number one and, number four is at 323, which is roughly 90 degrees aware from number one and three.

So, I only get one sample kind of in the other two-thirds of the entire vessel over the four years.

MR. SISK: I understand your --

MEMBER STETKAR: And, I don't -- it's fine if I expect the highest fluence to be in that area, I have it. But, I -- and, if it's intentional, I'm just curious why the highest fluence would be expected to be in that area?

CHAIRMAN BALLINGER: I mean, I'm looking at Table 5.3.7 here.

MEMBER STETKAR: Yes.

CHAIRMAN BALLINGER: And, there's -- when it says target fluence, there's only one capsule, D, which is the 323 at end of life which would be probably 40 or 50 EFPH.

MEMBER STETKAR: That's their nominal maximum targets.

CHAIRMAN BALLINGER: The EOL, right?

MEMBER STETKAR: Right.

CHAIRMAN BALLINGER: And, that's 6.44 times 10^{19}. And, so, that's the end of life target

fluence. Maybe that's the highest fluence at the end of the life.

MEMBER RICCARDELLA: As I look at it, all six of the capsules are in the regions of high fluence.

CHAIRMAN BALLINGER: Yes, oh, yes.

MEMBER RICCARDELLA: They're all in the peak area.

CHAIRMAN BALLINGER: Yes.

MEMBER RICCARDELLA: And, I think it's somewhat random whether you pick them. They might be trying to duplicate --

MEMBER STETKAR: It was just curious to me why, if I was going to sample four, I wouldn't sample them from axis symmetric locations around the vessel. You know, why --

MR. Y. KIM: Yes, this is Yunho Kim.

MEMBER STETKAR: You have to -- if you're going to answer, you have to come up to the microphone.

MR. Y. KIM: This is Yunho Kim.

Usually, we have a loading pattern, impure loading pattern. We designed that, for each quadrant at the same angle, they have the same fluence. Right?

MEMBER STETKAR: Should be, I would think so.

MR. Y. KIM: Yes, so, for, I think that the lesser fluence should add to it. The highest fluence point from each quadrant.

But, we don't have to select for each quadrant in sequence, you don't have to because in -- for each quadrant, you have the symmetrical --

MEMBER RICCARDELLA: Are you using it? Are you using low leakage core design?

MR. Y. KIM: Yes. We have a low leakage loading pattern.

Did that help you?

MEMBER STETKAR: I mean, in principle, it shouldn't make any difference. In principle, it shouldn't make any difference. It's just a curiosity, if it doesn't make any difference, why don't you pull samples from symmetric locations to confirm that it didn't make any difference?

MEMBER RICCARDELLA: In three of the four. Three of the four in symmetric locations.

MEMBER STETKAR: No, three of the -- I'm sorry, three of the four are within roughly a one-third of the vessel, 217, 224 and 323 on your figure is where three of the four come from.

MEMBER RICCARDELLA: But, they're also pulling 37, right?

MEMBER STETKAR: They are pulling --

MEMBER RICCARDELLA: Yes, 37 is --

MEMBER STETKAR: Yes, they're pulling 37, but three of the four -- they're not pulling anything from the --

MEMBER RICCARDELLA: The upper left quadrant.

MEMBER STETKAR: -- the upper left.

MEMBER RICCARDELLA: There might be a -- we need to -- this is --

MEMBER STETKAR: That's why I asked if there a particular reason --

MEMBER RICCARDELLA: Yes.

MEMBER STETKAR: -- why they're focusing on that area. If there isn't then, then okay.

MEMBER REMPE: Have you seen other ones and do they always -- for other plants, have they rotated around or do they use the data from the second one because they're doing calculation and they're trying to say, well, I want a nearby location to see how good my model is predicting?

I don't know how --

MEMBER STETKAR: I don't know.

MEMBER REMPE: -- that's what I was --

MEMBER STETKAR: I don't know. I don't know, it's just -- I'm sorry. I'm almost sorry I brought it up, but I was just really curious.

MR. M. KIM: Can I proceed?

There's a list of parameters of the coolant pump. Before, I indicated RCPs are vertical single stage bottom suction horizontal discharge.

This is on the motor shaft provides rotating the shaft for cooldown.

This figure shows the pump and the rotating parts.

It is supplied and lock to its shaft, pump alignment is maintained by a water lubricated bearing. These other bearings is water lubricated and the radial thrust bearings in the motor shaft, these are oil lubricated.

Pump and motor is collected by a coupling.

And, in the April meeting, there was a question about the inspections, motor inspections. For the rotor inspections, the motor has to be removed and the thrust bearing has to be removed,

motor stand also removed.

For seal changes, remove the coupling and the reason the coupling is on, and thrust bearing is slightly lifted and then some space is here around the rigid coupling. And, the seal assembly can be -- can put outside for inspection or change.

There was another question for high pressure cooler. They are normally cooled by two ways in the system. One is the seal injection from CVCS and number two is the high pressure cooler.

The seal injection water is combined with a recirculation flow at the jet pump and is injected to the seals through the high pressure cooler.

Note that the recirculation flow, not only the emergency condition when injection is lost, but during normal operation, the recirculation of flow is required. And, the CCW cools the separation flow.

Another question was the seal -- seal design is different from the existing plant. The function and geometry is the same but the motors are improved.

For the long-term exposure up to high

temperature and pressure environment for the -- I'm saying the difference is more clear. The difference between the previous plants and the APR1400 and this application design.

And, a test was done and the test results demonstrated the ability of the seal to survive an extended SEVU event.

There is an open item for the seal -- regarding injection and seals.

Section Number 5.4, two, Part I was to clarify regarding what condition during a 30 minute period. Thirty minutes in low flow CCW.

ICP seal injection is supplied by one of the two charging pumps and when the charging pumps are not available, the auxiliary charging pump can supply the steam injection to the ICPs.

And, in the response KHMP provided a description that the charging pumps and the auxiliary charging pumps are not directly impacted by a low flow CCW because the charging pump motor is air cooled.

MEMBER STETKAR: I'm sorry, the auxiliary charging pump motor is air cooled, is that correct?

MR. M. KIM: That is correct.

MEMBER STETKAR: Does the centrifugal charging pump require component cooling water?

MR. M. KIM: In this design, ARP1400 DCD application and air cooled motor is supplied and the charging pump has -- the charging pump mini-flow heat exchanger and the CCW is required.

But, the (inaudible) is served, it was not affected by the loss of CCW. The effects of the low CCW is not directly affected to the pump operation.

MEMBER STETKAR: So, you will have, if I lose all CCW the centrifugal charging pumps will continue to operate with no problem forever, is that true?

MR. M. KIM: The point is --

MEMBER STETKAR: No, I'm asking you a specific question. If I lose CCW, will the centrifugal charging pumps continue to operate with no problem forever? And, what I mean forever, I mean at least 24 hours.

MR. T. KIM: Yes, this is Tea Han Kim from KEPCO E&C.

Yes, we can run --

MEMBER STETKAR: Thank you.

MR. T. KIM: -- charging pump.

MEMBER STETKAR: We have that on the record.

Now, so, I understand if I lose CCW, I still have seal injection flow from the centrifugal charging pumps. So, I do not interrupt seal injection flow.

You now say that the reactor coolant pump can continue to run for at least 30 minutes with no damage.

MR. M. KIM: Ten minutes.

MEMBER STETKAR: I'm sorry, this says 30 minutes.

MR. M. KIM: Thirty minutes means the no impact.

MEMBER STETKAR: Well, okay. Then, where's the ten minutes? The ten minutes --

MR. M. KIM: It's not in the DCD --

MEMBER STETKAR: It's not in the DCD, is it? Where is -- what does -- you said ten minutes, so I want to understand what the ten minutes is.

MR. M. KIM: The difference is --

MEMBER STETKAR: If I -- so, let's now talk -- I lose all component cooling water, seal injection remains available because it stays

available. How much time do the operators have before they must trip the reactor coolant pump before I have damage to either the pump or its seals?

MR. M. KIM: The seal is okay. The bearing -- the pump and motor are oil bearing.

MEMBER STETKAR: Okay.

MR. M. KIM: Yes, it's limited, it has a limit of 30 minutes to affect.

MEMBER STETKAR: Okay.

MR. M. KIM: It is described in Section 5.012.

MEMBER STETKAR: Yes, yes, I read that.

So, that's 30 minutes. So, before it basically seizes. But, you said something about ten minutes just now, so what's the ten minutes?

MR. M. KIM: Ten minutes means if -- before ten minutes, the CCW is restored, there is any -- there is no damage to --

MEMBER STETKAR: Right, okay. But, I don't restore CCW in ten minutes. CCW never comes back, it's gone forever, what's the ten minutes then? What happens after ten minutes?

MR. M. KIM: The bearing may be damaged, but there is no effect to on the crossed

down characteristic.

MEMBER STETKAR: What about the seals?

MR. M. KIM: The seal is okay. The seal is okay for indefinite time, forever.

MEMBER STETKAR: Okay. I've asked many pump manufacturers this question and they always have told me they do not know how long the seals will last once the pump starts to vibrate significantly. And, I'm talking now about mechanical damage to the seals because the clearances are typically not very large.

So, as I start to get shaft vibrations, I can get rubbing on those intermediate seals and they can start to leak.

MR. M. KIM: What is the exact conditions? Seal injection is available or not?

MEMBER STETKAR: Seal -- even with seal injection available, I've asked other pump manufacturers this question and they can never give me an answer to what -- some of them say maybe five minutes, some of them say maybe ten minutes.

But, they've never been able to tell me much longer than that period that they will assert, I don't want to use guarantee because they never use that word. But, they will say that I will not

have seal damage.

And, I'm talking now mechanical damage, not hydraulic or thermal damage.

MR. M. KIM: So, if I just mentioned it contained in the DCD, 30 minutes is -- operational, 30 minutes is possible if the seal injection is available.

MEMBER STETKAR: Okay, let me --

CHAIRMAN BALLINGER: I think what he's saying is, let me -- the pump is running.

MEMBER STETKAR: Yes.

CHAIRMAN BALLINGER: Something happens to the pump.

MEMBER STETKAR: Nope.

CHAIRMAN BALLINGER: Which causes the imbalance of some kind.

MEMBER STETKAR: Nope.

CHAIRMAN BALLINGER: No?

MEMBER STETKAR: The component cooling water goes away. I stop cooling every single bearing on that pump. I've stopped cooling the oil --

CHAIRMAN BALLINGER: Right.

MEMBER STETKAR: -- I've stopped -- okay.

CHAIRMAN BALLINGER: So, the bearing dies.

MEMBER STETKAR: The bearing starts to -- the oil starts to heat up for the bearing.

MR. M. KIM: The temperature bearing and recoil is increasing.

MEMBER STETKAR: But, now, on most things, before the bearings actually fail, I have experience from this on a main turbine, you start to get very increased vibrations because of the viscosity of the oil gets pretty thing.

And, as the vibrations increase, there may be a chance that you can have mechanical damage to the seals.

MR. M. KIM: The APR1400 ICP has its reference, the -- we have the OPR1000 plans experience. And, the OPR1000 has referenced the system. And, they conducted an analysis and tests and three and four, conducted the full scale test.

MEMBER STETKAR: Okay. Well, when you say test, did they actually cut off all cooling water with a pump operating at rated reactor coolant system temperature and pressure with seal injection available to the seals and let the pump run for 30 minutes and confirm that there was no

damage? Did they do that? Was that the test?

MR. M. KIM: Yes, that is discussed in DCD 5412.

MEMBER STETKAR: I'm sorry, I read the DCD and the DCD is very careful to say that the bearing damage after 30 minutes would not be so bad that you would seize the pump shaft.

MR. M. KIM: Yes, that's right.

MEMBER STETKAR: Before the pump shaft seizes, I'm going to have very high pump shaft vibration at some time before the pump shaft seizes, right? Or no? You have no pump shaft vibration?

MR. M. KIM: I didn't bring the test results here, but as I know, as far as I know, the limit is the bearing temperature, not the vibration for this test.

MEMBER STETKAR: I'm asking -- you know, we may be talking about different criteria here. I'm talking about how long can I let that pump run with no component cooling water with seal injection available before the operators must trip the pump because the seals can be damaged. And, if that's 30 minutes, that's fine, I understand that, but I would like to know what test has been

performed to confirm that.

Because I have asked that same question to several different other pump manufacturers over the last 25 years of my life and none of them know the answer to that question.

MR. M. KIM: Yes, and I should check the test results available and then --

MEMBER STETKAR: Yes.

MR. M. KIM: I hope then, tomorrow morning, I --

MEMBER STETKAR: That would be great. I mean, if you -- because that was my second question. If you actually have run tests on this pump to confirm that, that would be very, very, very useful.

CHAIRMAN BALLINGER: That's an action item.

MEMBER STETKAR: But, it has to be a test at rated system temperature and pressure to ensure -- because I'm looking for seal damage. I'm not looking for seizure of the shaft. I'm not looking for bearing damage that's so bad that the pump -- the reactor coolant pump is somehow other damaged. I'm strictly looking at possible mechanical damage to the seals.

MR. SISK: So, Rob Sisk, Westinghouse.

Just, again, clarification on what you just said. You're not raising a safety concern, you're looking at it from more of equipment reliability or equipment --

MEMBER STETKAR: No, heck, I'm raising a reactor coolant pump seal LOCA concern and what is the time window available for the operators to manually trip the reactor coolant pumps after loss of component cooling water which is a risk assessment question and it's also a potential seal LOCA question. So, that's what I'm asking about.

Because there's a time window and any other pump manufacturer I've ever asked the question has come up with maybe ten minutes which was why I was curious about when you said ten minutes.

Some other pump manufacturers wouldn't tell me any more than five minutes.

MR. M. KIM: I believe that the -- during the 30 minutes, the temperature limit was not -- the test is under the limit and there was no --

MEMBER STETKAR: I mean, that would be -- in principle, you could do that. If you have a

huge oil reservoir with really good heat transfer, I can see how it can be done. I've just never seen it yet, so that's why I'm asking you if you have test available, it'd be great.

MR. M. KIM: I cannot show the test results here.

MEMBER STETKAR: Yes, I mean I understand, everything is proprietary, but, just make sure that we get the relevant information.

MR. M. KIM: I check and make an answer tomorrow.

MEMBER STETKAR: Thank you, thank you, thank you.

MEMBER SUNSERI: I have a question on this slide regarding the auxiliary charging pump. And, the experience in the U.S. with positive displacement charging pumps is that they are not very reliable, seal failures, high vibration.

And, as a consequence of that, many plants in the United States have either eliminated the use of positive displacement charging pumps or significantly restrict their usage time so that they can be available, you know, at the moment they're absolutely needed.

So, my question for you is, what is the

basis for having a positive displacement auxiliary charging pump and what benefit do you expect to gain from that?

MR. T. KIM: This is Tae Han Kim from KEPCO E&C.

The first, we use the is put in the auxiliary charging pump, the main purpose of that one is the we have to utilize that one for hydro test for that is a very high pressure. So, that kind of high pressure can be obtain easily from that kind of pump.

MEMBER SUNSERI: Right.

MR. T. KIM: So, that is one of the main purpose we're using that. And, also, we're using the charging -- main charging pump as a centrifuge, so there's another type.

And, also, this -- the auxiliary charging pumps, so it's not already running so it is for emergency case also we can use some limited time for operation.

MEMBER SUNSERI: Okay. So, I understand you said two basic reasons.

One is, it provides the head necessary to do the hydro which I understand that. U.S. plants bring in a temporary pump to do that since

it's done only once every ten years or so.

And, then, the other reason is because of diversity. You want something different than a charging -- a centrifugal charging pump.

Is that -- did I understand that correctly?

MR. T. KIM: Yes.

MEMBER SUNSERI: Okay, thank you.

MEMBER STETKAR: There are plants, I was surprise when I started working in Switzerland, this was, again, one plant that I've worked at uses only centrifugal, or I'm sorry, only positive displacement charging pumps and they don't -- they haven't had any problems with them.

I mean, it's a different -- apparently, some people make really good positive displacement pumps.

MEMBER SUNSERI: They're probably built like a Swiss watch.

MEMBER STETKAR: Yes, they --

MEMBER SUNSERI: We can make it work, but the effort it took to do that was not worth the benefit.

MEMBER STETKAR: No, it's -- but, I know that there are plants who routinely do use

positive displacement, you know, 24/7, 365 days a year for charging and, you know, haven't had the problems that we've had at some plants in the U.S. with them.

MR. M. KIM: Okay, the second part was regarding the seal test results discussed in the previous slide.

In the response, the seal test available to staff audit and the general summary of the test is incorporated into DCD Section 5.4.12.

Finally, on the chapter of motor and, as you can see in the figure, it has cylindrical shrink feet of the outer to the hull and then cortical shrink feet between the hull and the shaft.

The fly motor is 26 meter chrome molybdenum vanadium 14-5, a German motor. And, the first toughness K1c value is higher than 165 nanopascal to meter or 150 KSI or two-inch.

The technical report fly integrity provides a description for the flywheel design.

There are six open items for flywheel and these are follow up questions for the question number one through six.

The open items are question number 7

through 12.

To describe objectable approaching for determining fracture toughness of the fly motor in the response to this question, we specified the directness of determining fracture toughness in the test is being done and the test report will be available by November.

Question number eight is to clarify operating experience table. In the previous response to the previous question, there were some extra items.

For example, there was some spells not relevant to the operating experience. So, KHMP revised the operating experience table with the same tier and used the Korean and German plants and, of course, including other countries.

Question number nine, to apply ACP fly's test limit or one sort of the yield strength of the motor will provide technical justification for use of one-third of ultimate strength. In the response, the ACP fly stress limit of one-third of the yield strength of the material according to the SRP and the -- it depends on the test result of both the yield strength value and the fracture toughness data.

Question number ten, to revise the technical report include on analysis of hub and provide appropriate fatigue crack growth rates.

The technical report will be revised to add a separate stress probe of the hub.

Question number 11, to specify maximum flaw size as the acceptance criteria. So, in the response to this question, revised DCD subsystem to include the inspection as criteria.

Number 12, test and inspection proposed for the fly also applies to the hub.

In the picture, the hub has oil channels that will make it difficult to preform ultrasonic inspection flaw, but flaw propagation will be show as negligible.

As addition defense in depth notation, on PT or NT is proposed in the response. PT or OMT on the hub.

If you don't have questions --

MEMBER RICCARDELLA: Yes, I actually have a question. I've actually done some flywheel burst analyses in my career.

One of the things that made it difficult to address in U.S. plants was the flywheel was keyed onto the shaft and the key

weight presented a stress concentration.

This figure that you showed doesn't show a key. Is there a key way in the flywheel?

MR. M. KIM: That was one of the questions in the previous RAI and there is no key in the actual design.

MEMBER RICCARDELLA: Good, thank you.

MR. M. KIM: I turn it over Mr. Choi.

MR. CHOI: Seognam Choi speaking.

This is a stream generators.

Two steam generators are designed into APR1400. This steam generator is a vertical U2 lead circulation type of heat exchanger with 13,102 tubes of alloy 690 thermal ETTD materials. And, with the integral economizer and 10 percent marginal, the heat transfer area for tube pluggings.

This has also three types of tube support such as horizontal grade, vertical and diagonal support.

And, this slide shows the steam generators of APR1400 shippings with Barca of United Arab Emirates.

We have only one obstacle remaining related to the steam generator program. The

question is the leakage, first, to change the operating number.

And, second, is the clarifications of inconsistencies of the proposed leakage barrier within technical specifications and Chapter 15 of DCD.

We submit the answer at June 16th this year and we believe this RAI will be acceptable because we correct the inconsistencies as requested.

MEMBER REMPE: So, just to make clear, if I read this new graph, it implies to me that, in Chapter 15, you did use .6 gallons per minute in your Chapter 15 analysis?

MR. CHOI: Yes.

MEMBER REMPE: Okay, good. Thank you.

CHAIRMAN BALLINGER: I have a sort of general question about the steam generator, especially the tubing.

I'm sure you're aware of what happened at San Onofre, their large steam generators like these, 690 tubing and they had a case where one plant had excessive wear and leakage eventually.

And, the other plant had the same tubing with nominally the same exact construction,

but didn't.

And, they traced that, at least if my memory serves, to slight differences in the way the surfaces were treated on the tubes and the clearance between the AVBs in the upper part of the section.

And, so, those steam generators were nominally manufactured in exactly the same way, but yet, that small difference made a huge -- had a huge effect.

Now, you -- in the DCD and everything, there's a fairly large discussion of that issue. But, I still come down to asking the question of, given the fact that -- how are we going to -- how are you going to be sure that that problem with that similar that happened to the San Onofre steam generators doesn't happen here, given the fact that just the small difference within a specification for fabricating the steam generators made a huge difference?

In other words, what --

MR. CHOI: I'm not sure the differences of San Onofre or our APR1400. And, then, the -- San Onofre manufacture ways that same manufacturer of the APR1400.

MEMBER BALLINGER: No, no, that's for sure. That's not the point. The point is, is that in spite of the fact that whoever manufactured the steam generators, we know who they were, they were manufactured to the same specifications, I presume.

And, yet, they had excessive wear in one and not excessive wear in the other. And, so, my question is -- and, these are in the upper bundle where the AVBs are, irrespective of what's down below with the egg crates and things like that. So, that's different.

But, how are we -- how are you going to assure that that problem doesn't -- is not -- well, very unlikely to exist in these steam generators? Because they are roughly the same size, same tubing.

I don't know what the ABV material is, the egg crate material is a 409 or something stainless steel. But, I don't know what the -- I forget what the AVB material is.

MEMBER RICCARDELLA: Let me just, I mean, the problem was fluid elastic instability.

CHAIRMAN BALLINGER: Yes.

MEMBER RICCARDELLA: And, what could you tell us to give us some confidence that we

don't have a fluid elastic instability problem in these steam generators which are of --

I mean, I know there's a couple of paragraphs in the DCD about it, but, in terms of testing and experience, I mean, are these similar to the OPR1000 steam generators? Are they larger? Just some discussion of that would be helpful.

MR. CHOI: Yes, there is a two improvement of the steam generator design.

The first one is that we designed it going central regions and there's high flow regions and we blocked in the flow.

The egg crate flow, this plate is added center of regions.

And, the second one is the vertical horizontal central lines have the one more -- or one or two more vertical supports added. And, that is the main improvement of the APR1000.

CHAIRMAN BALLINGER: That's right.

MR. CHOI: Yes.

CHAIRMAN BALLINGER: But, according -- the analysis that you're doing is a computer analysis to predict fluid elastic instability. They did the same thing, I think, for the other steam generators and the CFD people will claim that

they can cure pattern baldness with their CFD code, but, the fact of the matter is, they can't.

MEMBER RICCARDELLA: Well, but, I mean there are successful steam generators of this size operating. Palo Verde's a case in point.

I think, you know, it comes down to operating experience and test data.

CHAIRMAN BALLINGER: So, again, it's back to the test data. So, the design code, the rules will say that if you suspect that you're going to have an issue, you need to confirm that you don't.

So, my question is, how are you going to confirm that you don't?

MR. CHOI: As I said before, that there's two improvement of the support and control the chemicals and then I think so.

CHAIRMAN BALLINGER: Okay. So, you're saying that those improvements are going to fix --

MR. CHOI: Yes.

CHAIRMAN BALLINGER: -- they're going to ensure against the problem?

MEMBER SUNSERI: Let me ask the question a little differently.

The codes you're using to design the

parameters inside the steam generators, the flows, the thermal hydraulic characteristics, do you have sufficient operating experience with those codes or are any of the codes new and unique to this design and being used for the first time?

MR. CHOI: Yes, and we use the same EPRI code at those three.

MEMBER SUNSERI: You said EPRI codes?

MR. CHOI: Yes.

MEMBER SUNSERI: Okay.

CHAIRMAN BALLINGER: Thank you.

MEMBER RICCARDELLA: How much different are these steam generators from the OPR1000 steam generators?

MR. CHOI: Size?

MEMBER RICCARDELLA: Size, yes.

MR. CHOI: Yes, well, OPR is about -- total is 821 inches high and then this APR1400 is 905 inches tall.

MEMBER RICCARDELLA: A 100 inches taller?

MR. CHOI: Yes. And, it's a bigger --

MEMBER RICCARDELLA: Bigger diameter?

MR. CHOI: Yes, the diameter is.

MEMBER RICCARDELLA: Yes, you mentioned

-- you didn't mention in the DCD that the tube support structure, the egg crate structure is the same?

MR. CHOI: Yes.

MEMBER REMPE: I'm sorry, maybe -- I know you asked about APR, but Shin-Kori's steam generators, are they the same as this plant's?

MR. CHOI: I'm sorry. Yes, there is increase the tube support over three maybe.

MEMBER RICCARDELLA: Okay, so --

MR. CHOI: It goes with these allow the tube size.

MEMBER REMPE: Okay. So, this will be different than Shin-Kori?

MR. CHOI: Yes.

MEMBER RICCARDELLA: Than Shin-Kori 3?

MR. CHOI: It's the same.

MEMBER RICCARDELLA: This is Shin-Kori 3?

MEMBER REMPE: It's the same as Shin-Kori 3, that's what I meant to say, the newer plants, so it's the same. And, you've manufactured those with Doosan --

MR. CHOI: Yes.

MEMBER REMPE: -- manufactures them?

MR. CHOI: Yes.

MEMBER REMPE: Okay.

MEMBER RICCARDELLA: So, if there's a problem like San Onofre, we'll see it before -- right?

MEMBER SUNSERI: So (inaudible) three back around.

CHAIRMAN BALLINGER: It's all the fault of the metallurgist.

MEMBER RICCARDELLA: It's what?

CHAIRMAN BALLINGER: It's all the metallurgist's fault.

MEMBER RICCARDELLA: Well, I mean, you know, if you're operating close to an instability point, you can see those kind of differences. If you're operating, you know, well below the instability point, then the small differences probably won't make that much of a difference.

CHAIRMAN BALLINGER: But, if you don't know that's you're close to the instability point --

MEMBER RICCARDELLA: Well, but I mean they've done analysis and testing to say that they're not close -- that they're well below the instability point.

CHAIRMAN BALLINGER: I thought they said they've done analysis, but I didn't read that they'd done testing. Okay.

MR. T. KIM: Yes, 5.4.7 shutdown cooling system. Shutdown cooling system is used to reduce the temperature of the ICS from a hot shutdown condition to refueling condition.

One of the design features of the APR1400 is the shutdown cooling pumps and the containment spray pumps are identical and functionally interchangeable.

MEMBER STETKAR: Let me ask you a couple of questions about this system before you go to the RAI.

If I look at the drawings in the DCD, in particular, Figures 5.4.7-3 which shows the shutdown cooling system as shown here, or Figure 6.3.2-1 which shows more detail of safety injection and shutdown cooling and Figure 6.2.2-1 which shows the containment spray system, I do not see any manual isolation valve on the suction side of the containment spray valve.

And, your drawings are pretty good about showing valves and flanges and reducers.

So, if I must remove a containment

spray pump from service, it seems to me that I must close the suction valve from the IRWST for that pump which prevents me from using the shutdown cooling pump in that train.

So, I'm curious about how do I align a shutdown cooling pump to replace a containment spray pump? What values -- what particular valves are open to do that? Or is a manual valve missing from your drawings?

MR. T. KIM: This is Tae Han Kim from KEPCO E&C again.

Actually, we have a cross connector line between the shutdown cooling system and the shutdown containment spray system. But, the DCD drawing is not showing everything in details, but we have some connecting lines.

MEMBER STETKAR: Well, it's -- I, you know, I see the connecting line. I understand how the pipes are connected. And, your drawing, in particular, for the containment spray system and the safety injection shutdown cooling system are rather detailed.

They're so detailed for the spray system is you show a 12-inch to 14-inch piping reducer. It shows a locked open manual valve V1003

on the discharge side of the pump.

I see no manual valve on the suction side of the containment spray pump. And, I cannot find one in any drawing.

So, if there's no valve there, if that's the certified design with no valve there, then, if I must remove a containment spray pump from service to repair it for whatever reason, it seems that I must close the containment spray pump suction valve from the IRWST because that is the only isolation valve that I have for the pump.

If I close the spray pump suction valve from the IRWST, that is the same suction supply to the SCS pump, meaning, I cannot use the SCS pump.

Do you understand what I'm saying?

MR. T. KIM: The first question, the cross connecting -- one of the cross connecting valve is 340 --

MEMBER STETKAR: Let me give specific valve numbers.

MR. T. KIM: Three hundred and forty.

MEMBER STETKAR: Three forty, yes, 340 is --

MR. T. KIM: Is the correct one of --

MEMBER STETKAR: Right.

MR. T. KIM: -- the connecting valves.

MEMBER STETKAR: Right. But, if you look at the flow diagram, I must close 347, 3-4-7, to isolate the suction of the spray pump because there is no manual valve between the junction to 340 and the suction of the spray pump. I cannot find a manual valve in your certified design.

So, to isolate a spray pump, it seems I must close 347. If I close 347, that isolates 340 and the shutdown cooling pump cannot take a suction from the IRWST.

MR. T. KIM: Actually, right now, we didn't bring the whole detail P&ID here so --

MEMBER STETKAR: Okay. So, maybe something is missing from the certified -- the drawings and the certified design. But, I get really upset when things are missing from the drawings and the certified design because they are the certified design.

And, indeed, the drawings do show a large number of manual valves.

So, I'll ask you to check to see, is there a manual valve in the line between the junction of valve 340 and the suction of the containment spray pump, please?

And, if there is, I'm good. I understand how it works. I can get water there a different way, but your whole discussion says that I use that cross connect with 347 and whatever the other one is there that I can't read, 341, I think.

So, that's a question I have about that valve.

There is one, you can see, in the suction line of the shutdown cooling pump going the other way, it's V-1-0, my eyes are not very good, it's 5 or 6.

Different, but related topic, you said that, if I use a shutdown cooling pump to replace the spray pump for the containment spray function, I line it up so I have suction discharge. And, it says that the -- in that condition, the shutdown cooling pump is capable of being automatically started by a safety injection signal or a containment spray signal.

Do the operators make that transfer? In other words, how do the shutdown cooling pumps receive the automatic start signal from containment spray or safety injection? How is that automatic and a start signal enabled?

MR. T. KIM: When the operator line up

the SCP, is there a CSP with the manual valve in the cross align, the interlock for this valve enable the SIS and the CSA containment spray injection system go the pump.

MEMBER STETKAR: It's interlocked with the valve?

MR. T. KIM: Yes.

MEMBER STETKAR: That's exactly what I was looking for. Thank you, thank you on that one. Thank you. I was worried about starting the pump without the valves open and, if it's interlocked with the valve, that's great. Thank you. But, still look for that manual valve on the suction of the spray pump.

MR. T. KIM: Actually, this drawing, I might can explain, but, you know, I have to define each bandwidth here, so I will have another chance to talk with this drawing.

MEMBER STETKAR: Okay, thank you.

MR. M. KIM: Go to the next slide.

The SCS has one open item. Question number four, to revise the ITAAC for gas operation addressing those things related to shutdown cooling system, safety injection system, containment spray system.

In the response, KHMP revised the ITAAC to include the as-built evaluation for gas containment.

MEMBER STETKAR: Before we go to the pressurizer, where -- there is the suction line, the SCS suction line from the hot leg located? Is it at the bottom of the hot leg or is it up -- where relative to the mid plain of the hot leg is the actual suction line located?

MR. T. KIM: This is Te Ha Kim, again, KEPCO E&C.

It's the bottom line.

MEMBER STETKAR: It's directly from the bottom?

MR. T. KIM: Yes.

MEMBER STETKAR: Thank you. Good.

MR. CHOI: Pressurizer, Seognam Choi speaking.

The function of the pressurizer is to control the rate of coolant system pressure and the volumes by the operational transient without opening the safety valve which the over pressurization arises is protected by the full pilot operating safety relief valve is on the top of the pressurizers.

The inside volume of the pressurizer is 2400 cubic feet and this is connected to the third line at the bottom.

This is the design parameters of pressurizer for APR1400 and the shapes.

And, this figure is APR1400 pressurizer related to the hydrostatic testing at the shafts.

MEMBER STETKAR: I have -- I'm sorry, I have one more question on shutdown cooling system, and you don't need to go back on slides.

In Section 5.4.7.2.2 of the DCD, there's a statement that said the failure position of each valve on loss of actuating signal or power supply is selected to provide reasonable assurance of safe operation.

System redundancy is considered one defining the failure position of any given valve.

In Section 5.7.2.5, you note that the SCS does not utilize any pneumatically operated valves. So, the only valves that I could see on your flow diagrams are either manual valves or motor operated valves.

In my experience, most motor operated valves, if I do not have any power to them, they fail, so-called, as-is. If it's closed, it remains

closed. If it's open, it remains open. And, that's also typically the case if I don't have any signal for it.

So, I'm curious about this statement that you have selected, fail positions for the valves in the SCS and that those failed positions have been defined based on some consideration of redundancy of the system.

Are there -- let me ask the question more simply, are there any valves in that system that do not fail as I would call it, as-is. If it's closed, it remains closed. If it's open, it remains open.

MR. T. KIM: Yes, this is Hae Han Kim again.

As you explained, sir, MOVs fail as-is, but some of the SOV is.

MEMBER STETKAR: But, I didn't see any SOVs in the SCS. I saw them in other systems. Are there SOVs in the shutdown cooling system?

MR. T. KIM: Sorry, I misunderstood with that.

MEMBER STETKAR: Oh, are there or are there not in -- this is the only place that I found in Chapter 5 where it discussed failure positions

where I did not see any SOVs.

So, I was curious why this statement applies to the shutdown cooling system?

MR. T. KIM: I'm sorry, actually, I misunderstand the Chapter 6 from --

MEMBER STETKAR: Oh, okay.

MR. T. KIM: -- so, that's the one, I think.

MEMBER STETKAR: Let me just put it on the record, I -- if it's only motor operated valves and they fail as-is, I understand how the system works, I just don't know what, in particular, that statement that I quoted means in the context of the certified design.

Because, to me, it implies that there's something active that was done to perhaps determine that a valve would reposition itself.

MR. SISK: Okay, Rob Sisk, Westinghouse.

I understand the comment. I think you're looking for automatic adjustments for repositioning. But, it may be simply that the system is aligned for this safe performance --

MEMBER STETKAR: And, I --

MR. SISK: -- if the valves do not

operate.

MEMBER STETKAR: I'm pretty happy with that statement, Rob. I trip over things where you talk about failure positions being selected as if --

I've seen people do things, for example, if a signal is not available, but power is available, they'll reposition a valve to what its so-called failure, you know, best guess in terms of the right position should be.

So, I've seen people get pretty creative with those types of things. You can't do much for a motor if there's no power to it. But, if power is to it and you don't have a signal, there might be something smart that says, oh, go open that valve that used to be closed because that's where I'd like it.

MR. SISK: I understand the comment. We'll have to discuss that internally.

MEMBER STETKAR: Yes. I doubt that that's the case, but I just stumbled over that and it gave ma lot of pause.

MR. SISK: Understood.

MR. M. KIM: 5.4.12, reactor coolant system high point vents.

Reactor coolant gas vent system, the function is to discharge non-condensable gases and steam from the high points of the RCS for venting or pressure controlled during post-accident conditions.

The IRWST collects the -- collects and condenses the steam discharge from the coolant gas vent system.

MEMBER STETKAR: Okay. I had a really difficult time looking at the DCD, several sections of the DCD to understand what the real connections are to the IRWST.

There are different simplified drawings in the DCD that show different connections and they're all very simplified. Is there any drawing anywhere in DCD Tier 1 or Tier 2 that unambiguously shows every flow path into the IRWST, out of the IRWST and things like the locations of spargers within the IRWST?

I couldn't find one, but I admit, I did not read all -- every page of the DCD.

MR. SISK: So, again, Rob Sisk, Westinghouse.

For clarification, you're looking for all of -- a drawing that shows all of the lines

into the IRWST and the spargers?

MEMBER STETKAR: Yes. It's, in simplified, in a way that is unambiguous. Because, for example, in one drawing, I'm led to believe that the POSRVs -- let me use RCGVS as an example.

In one drawing, I'm led to believe that RCGVS discharges through its own line. In another drawing, it seems to discharge in a line that comes from the containment purge system.

So, I don't know how many discharge lines are, I don't know how they're oriented. And, because I can't see anything, I don't know if there are any valves in those lines that can be left closed.

In particular, other than the one drawing that I mentioned earlier this afternoon, or this morning, no other drawing show those two motor operated three-way valves in the POSRV lines.

So, without kind of an unambiguous complete drawing, it's really difficult to see how the thing is piped up.

MR. SISK: But, it sounds like you're looking -- Rob Sisk, Westinghouse again.

It sounds like you're looking for basically the system drawing. Because, if you're

looking just at the IRWST, you're not going to see the --

MEMBER STETKAR: No, that's right. But, I've looked at the system drawings and the system drawings tend to just show arrows. And, I went to the IRWST drawing in Chapter 6 and, in particular, that drawing doesn't even show the POSRVs coming into to the IRWST. It just shows some other lines.

So, in Chapter 6, the one that ostensibly shows the IRWST, and I think there was a -- I think there was an RAI about that and the response to that RAI said, well, yes, in Chapter 6, we just showed the IRWST. It doesn't necessarily show every line going into it.

I couldn't find one drawing anywhere that showed everything.

MR. SISK: It would be complicated drawing. But, I -- we understand what you -- your comment.

MEMBER STETKAR: Well, it -- and, in terms of complexity, I'm more concerned about how lines are plumbed together, piped together and if there might be valves in those lines and things --

People typically will show the location

of the spargers. For example, I don't know, are the spargers relatively high in the IRWST? Are they low? Where are the spargers in relation to the suction lines for the SI pumps and the spray pumps?

MR. SISK: Let me ask Mr. Tae Han Kim if he'd like to comment.

MR. T. KIM: I believe if you check the Figure 6.8-3.

MEMBER STETKAR: 6.8-3? Yes.

MR. T. KIM: Yes.

MEMBER STETKAR: Yes.

MR. T. KIM: You checked already?

MEMBER STETKAR: It doesn't -- that one, in particular, does not show the discharge lines from the POSRVs or the RCGVS which is what started me on my hunt.

Figure 2.4.1-2 in DCD Tier 1, now, is a very, very simplified drawing. And, that leads me to believe that lines are connected together.

All the other system drawings just kind of show arrows going out to the IRWST.

MR. SISK: John, we're taking down the comment and we'll look into that.

MEMBER STETKAR: Thanks.

MR. M. KIM: Have you mentioned the

Figure 2.4.5-1 of Tier 1? The --

MEMBER STETKAR: I looked at 2.4.1-2. Which one did you say?

MR. M. KIM: 2.4.5-1 Tier 1 is this diagram is for RCGVS.

MEMBER STETKAR: 2.4.5-1? I did not look at that. I'll look at that later.

MR. SISK: Figure 6.8-3.

MEMBER STETKAR: Figure 6.8-3 does not show the POSRVs or RCGVS, does it?

MR. SISK: It's got a lot of stuff on it.

MEMBER STETKAR: It's got a lot of stuff on it, it doesn't have that. It's got more stuff than any other drawing I could find.

MR. M. KIM: Yes, every diagram is simplified.

MEMBER STETKAR: Every diagram is simplified and several are somewhat different.

MR. M. KIM: The figure in the response to RAI question 82, RAI 82-44 --

MEMBER STETKAR: Yes.

MR. M. KIM: -- question number 5.2.2-2 --

MEMBER STETKAR: That refers --

MR. M. KIM: -- the figure shows the IRWST and three-way valves and POSRV, but it is also a simplified one.

MEMBER STETKAR: Yes, it is. I looked at that one, too. Okay.

You only have this one slide on RCGVS, right? Okay.

MR. M. KIM: Yes, it is.

MEMBER STETKAR: Let me ask you a couple of questions about that.

In -- let me get my figure here first -- give me a second because we're -- you're ahead of me here.

If I look at Figure 5.4.12-1 which shows the RCGVS flow diagram, there are manual valves, V212, V2300 and V1430 that -- well, let me be very clear here because we're on the record.

Figure 5.4.12-1 shows V1430 in the discharge line from RCGVS to the IRWST. So, if that valve is closed, if that valve 1430 is closed, I cannot vent from the pressurizer or from the reactor vessel head to the IRWST, which is part of my question about understanding how things are plumbed together.

I had to go look in other places, if I

look at Figure 5.1.2-1, that shows the connection from the reactor vessel head to the RCGVS and it shows a manual valve, V212 in that line.

So, if that valve is closed, I cannot vent the head.

If I look at Figure 5.1.2-3, that shows the connection from the pressurizer to RCGVS and that drawing shows valve V2300 manual valve. And, if that valve is closed, I cannot vent the pressurizer.

So, I have a manual valve from the reactor vessel head. I have a manual valve from the pressurizer. And, then, I have a manual valve in the discharge line to the IRWST.

My first question is, do those valves have position indication in the main control room? And, you may not have that. I mean, you know, yes.

MR. SISK: We've taken a note down.

MEMBER STETKAR: Yes, okay. My follow up question on that was, if they don't have position indication in the main control room, when are those valves closed during normal plant shutdown and refueling operations? You know, when are they repositioned?

Because they had position indication

and, you know, the operators know that they're open, then I'm okay.

I just want to make sure that I understand the solenoid valves in the RCGVS, according to Table 5.4.12-1, those are DC powered valves, is that correct?

MR. OH: This is Andy Oh, KHMP, Washington Office.

You are asking about RCGVS solenoid valve power source?

MEMBER STETKAR: Yes, yes, there's a table, I think I know the answer, I'm just trying to confirm it.

MR. OH: Yes, they are DC power.

MEMBER STETKAR: They are DC power?

MR. OH: Mm-hm.

MEMBER STETKAR: Thank you.

Because, it was a bit confusing when I read it. I think I know the philosophy. The text in the DCD emphasizes the fact that RCGVS can receive power from the alternate AC generator. That's not quite accurate.

The alternate AC generator can provide power to either the battery charger in Division A or the battery charger in Division B. Those

batteries are rated for eight hours.

The Division C and Division D batteries are rated for 16 hours.

So, I need the AAC generator to be able to operate RCGVS for 16 hours. And, I kind of get that.

Now, my question is, and this is for the PRA people, you will not be able to answer it today, I just want to put it on the record.

According to the plant design, after 16 hours, I cannot operate RCGVS because the solenoid valves will fail closed. Even if I had the alternate AC generator, because that can only supply one division.

So, is there any part of the PRA, in particular, the Level 2 PRA, where you take credit for venting the pressurizer, in particular, through the RCGVS to keep pressure low in the Level 2 PRA after 16 hours?

Typically, in a Level 2 PRA, you'll ask a question of, is reactor coolant system pressure high or low and is it maintained low?

And, I don't know -- I didn't go look at the PRA to examine how they answered that question. I didn't have time. And, I don't know

what the timing is in the PRA.

So, this is more of a PRA question. I'll bring it up again in the PRA, but this is kind of a warning shot before we get to Chapter 19.

I just want to, first of all, I wanted to make sure that they were at DC power and you've answered that question. So, thank you.

How big is the pressurizer vent piping? The drawing, this Figure 5.4.12-1 indicates that the piping from the reactor vessel head is three-quarter inch diameter. And, there was a question about, I know that the staff asked the question about a LOCA in that line.

I couldn't find a size on the line from the pressurizer. Do you know how large that is?

I mean I can speculate, but I don't want to do that.

MR. J. KIM: Which line?

MEMBER STETKAR: It's the line -- it's the vent line from the pressurizer to RCGVS. So, it's that common vent line that comes off the four POSRV head. It's the line that goes over to the solenoid valves that are shown on Figure 5.4.12-1.

MR. OH: This is Andy Oh, KHMP Washington Office again.

And, from the pressurizer to the RCGVS, the size of pipe is two inch.

MEMBER STETKAR: Two inch? Okay, thanks. That's -- I know it was -- it's shown, it's two inches through the solenoid valves, but I didn't know if it was larger or smaller than two inches coming out. So, it's a two inch line? Thank you.

Okay, you're going to get to take a break, so I only have one more question here. Maybe two more.

In the DCD, there's a discussion about leak detection for RCGVS and I know how the system is configured.

It says the RCGVS is designed not to be used during normal operation. The piping is designed to slope downward to drain to the reactor drain tank and the IRWST.

The potential for leakage is very low and a leak detection system is not required.

I always get concerned when people say the potential is very low. I understand that the desire, in fact, is to open these lines. But, my concern is, if those solenoid -- if two of the solenoid valves are leaking or for some reason are

stuck open and the operators want to control a depressurization, they may not have the control that they want.

The system actually is configured for leakage detection in the same way that other systems that you discuss in the plant can discover leakage.

And, I was curious why you specifically say you don't need leakage detection on this system when you have the capability of detecting leakage by pressure transmitter PT-106, in particular, and the ability of the operators to open the valve to the reactor drain tank to check for amount of leakage flow if you detect high pressure.

So, it's -- personally, it's not a big concern from my perspective. This, again, is a subcommittee meeting, so I can say that.

But, I'm always curious about people saying I don't need something because something is very unlikely.

Typically, our experience then shows when that unlikely thing happens, people are very concerned.

So, I'd ask you to recheck that statement about why you don't need leakage

detection. Because you may have it already.

Now, the last question that I have -- actually, I'll wait and I'll ask the last -- I'll ask the staff my last question because it's more relevant to them.

So, I'm sorry, I'm done now.

MR. CHOI: Seognam Choi speaking.

Yes, these components support it.

This slide shows the schematic view of the APR1400 or main component in the support. The reactor vessel have the full operator columns and that is connected to the bottom of the floor in others.

The steam generator has a sliding base and sliding base is connected to the bottom and two keys are located at the upper shelf of steam generators.

Pressurizer support is integral support and this is at the bottom is connected to AE building. And, with coolant components four breaker columns at the bottom and two upper and lower rated columns and two reactor snivels at the bottom of ICD motor stem.

MR. M. KIM: Minseok Kim is presenting.

In summary, Chapter 5 describes the

design features of the coolant system of APR1400. All the ICS components are located inside the containment building.

The table shows the pre-installed parameters for the system. ICPD components are considered with the 10 CFR 50.2 as the depiction of the ICPD and 10 CFR 50.55(a) the code and standards.

The applicable code and standards of the components are listed in Chapter 3, Table 3.2-1.

Chapter 5 is on a success path for completion on schedule. An SER with open items was issued. A path for closing each open item has been discussed with the NRC staff and is being implemented.

KHMP believes that the open items, except for flywheel, which is in progress, have been addressed.

Chapter 5 as reviewed and marked up in response to NRC's RAI will be incorporated into the next revision of the DCD.

Monitoring of changes to other chapters for impacts on this chapter is conducted to maintain consistency.

This is all of our presentation we prepared. I thank you for your attention.

CHAIRMAN BALLINGER: Thank you.

Now, we have an option, we can --

MEMBER STETKAR: No, you're going to ask does anybody have any more questions first.

CHAIRMAN BALLINGER: Okay. That's all I was going to say, but it is what I'm going to say now. I was going to offer a break and then talk.

MEMBER STETKAR: All right. Because, the staff's going to come up, right?

CHAIRMAN BALLINGER: No.

MEMBER STETKAR: Oh.

CHAIRMAN BALLINGER: We've now succeeded in getting ourselves right back on schedule.

MEMBER STETKAR: Okay.

CHAIRMAN BALLINGER: A little late, actually.

MEMBER STETKAR: Let me, you know, let me cut to the chase.

Appendix 5A discusses interfacing system LOCA analyses. And, first of all, let me say that this is the first DCD where I've seen such a comprehensive assessment of interfacing system

LOCA pathway as I think that it was really -- I was really pleased to see that. I've not seen anybody address that issue in this level of completeness before. So, let me preface any comments that I have with that.

I did have a couple of questions related to those analyses.

The first question I had is simply one of completeness that you did a very comprehensive evaluation of every system connected to the reactor coolant system to identify the flow paths, isolation valves and so forth, except for one.

And, that is the auxiliary pressurizer spray line. Why did you not include that? Your conclusion, I'll tell you, I think, would be the same for the charging line as the auxiliary pressurizer spray line. But, you didn't include it.

So, when I went, conceptually, around the reactor coolant system and looked for piping, that one seemed to be missing.

So, that's just a comment. It's -- everything else seemed so complete that I didn't know why it wasn't there. I mean, you even went down the sample lines.

MR. SISK: We're going to have to take a look at that to see if there's a particular reason or if it's --

MEMBER STETKAR: Well, I don't know whether there's a reason or not.

MR. SISK: Yes, we'll --

MEMBER STETKAR: I'm not -- it was -- everything else was so complete that I was looking for something that was missing and I thought I found something.

A couple of questions, and I understand the analyses, I think they're really good. There are a couple of places where you take credit for manual operator actions to isolate systems before other systems become over pressurized.

And, in particular, I'm looking at the let down and charging lines where you say, well, the operators have pressure indications. There are relief valves to the reactor drain tank.

How large is -- and Dick Skillman asked this morning for a different reason -- what is the capacity of the reactor drain tank in terms of volume and what is its pressure rating?

Because, essentially, your analysis says I can blow to the reactor drain tank for long

enough so that the operators recognize the need to isolate the let down or charging line and stop the release. I mean, that's basically the philosophy for those two pathways.

So, if the -- for example, if the reactor drain tank could hold two liters and failed at a pressure of, you know, two pounds, that would not provide very much time for the operators to isolate the lines.

If it's a relatively large tank, that provides much more time. So, that's why I'm asking about that particular question.

MR. M. KIM: The internal volume of the reactor drain tank is specified in the Table 934-2.

MEMBER STETKAR: 93 --

MR. M. KIM: A thousand gallons and the minimum is added.

MEMBER STETKAR: I can look it up. It's 934-2, you said, the table number?

MR. M. KIM: The table number is 934-2.

MEMBER STETKAR: 934-2, I will look that up in the interest of time. Thank you.

CHAIRMAN BALLINGER: Now, I'm confused.

MEMBER STETKAR: Why?

CHAIRMAN BALLINGER: It's -- we're

finished in effect.

MEMBER STETKAR: No, we're not. I have one more question.

CHAIRMAN BALLINGER: Okay.

MEMBER STETKAR: As best as I can -- this, again, is just so I understand the integrated plant. As best as I can tell, each main steam safety valve has a rated capacity of slightly more than five percent rated power. Is that --

I think I found -- I did some calculations and I came out with each main steam safety valve having a rating capacity of about 5.3 percent full power.

And, that seems to be consistent with statements that all 20 MSSVs can relieve full power. Each POSRV, my guess is, can relieve about 3.5 percent full power. And, I'd like to confirm that my estimates are correct or, they don't have to be precise, but approximately correct.

The reason is, not for anything in Chapter 5, I'm looking out at Chapter 19 and the success criteria for the PRA. So, I want to make sure that I understand what those relief capacities are.

So, if you can confirm that. As I

said, I did a couple of back of the envelopes and I think I'm close, but I'm not sure.

MR. SISK: Again, we've got the comment, we'll take a look at that.

MEMBER STETKAR: Thanks.

And, now, I'm done. I'm finished, no more.

CHAIRMAN BALLINGER: I think I'll wait a few seconds just to see.

Thank you.

Okay, so, now we're, again, we have an option of taking a break and then coming back to have any further discussion or having discussion now.

So, my vote is, since we're already half an hour late, we might just as well go around the table and ask about questions.

MEMBER MARCH-LEUBA: I'll second that one.

CHAIRMAN BALLINGER: Soft vote.

Okay, any questions around -- from members? I think we have to open the line, right?

In the meantime, are there any questions from the audience? Any people want to make a statement or anything?

Hearing none, hopefully the line will be open.

Okay, I think -- okay, I think the line is open. Just to be sure, is anybody out there, can you make a human noise so that we can verify that?

MS. BANERJEE: Yes, Maitri's out there.

CHAIRMAN BALLINGER: Great, that's a human.

MEMBER POWERS: You're very generous in your appraisals here.

CHAIRMAN BALLINGER: Okay, is there anybody out there that would like to -- any member of the public out there that would like to make a statement?

Going once, going twice, closed.

Okay, so there are no comments. Can we close the line?

Okay, I think that pretty much concludes today's -- we did -- oh, now we can go around the table.

Joy?

MEMBER REMPE: I don't have any additional comments. Thanks for all your presentations.

MEMBER BROWN: Thank you very much. Charlie Brown. Thank you very much and who can compete with John. So, I give up.

MEMBER MARCH-LEUBA: I have no further comments.

MEMBER STETKAR: I'd like to thank you. You covered a lot of material today. I'm obviously a detail oriented person, but one of the roles that the ACRS provides that is to look at integrated plant safety.

So, a lot of the things that I ask, I'm trying to understand system interactions. I'm trying to understand how the design relates to success criteria and the risk assessment.

I'm trying to understand how the risk assessment results feed back into the design and so forth.

So, many of my questions come from that direction.

In the future, I'll just make this statement on the record, less emphasis on open RAIs and more emphasis on technical information from the design would be appreciated. We can read RAIs and things like that from the staff. So, from your perspective, keep it more technical.

But, again, thank you.

MEMBER SUNSERI: Thank you for your presentations. I enjoyed them and I have no further questions.

CHAIRMAN BALLINGER: Dr. Powers?

MEMBER POWERS: Nothing.

CHAIRMAN BALLINGER: Pete?

MEMBER RICCARDELLA: I'm impressed by the level of detail that you've gone into and particularly by how much you've recognized the lessons learned with existing plants and the problems with existing plants and how to avoid those, particularly in the materials area.

CHAIRMAN BALLINGER: And, I would like to thank you for a great -- a very good presentation.

I have in front of me from Maitri a set of action items and I would propose that tomorrow near the end of our talk those -- we should go through those action items to make sure everybody's on the same wave length. Because, we could do it now, but it would take a long time.

So, with that being said, I think we are adjourned until tomorrow morning at 8:30.

(Whereupon, the above-entitled matter

went off the record at 3:27 pm.)

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
APR 1400 Subcommittee Meeting

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Thursday, September 22, 2016

Work Order No.: NRC-2629 Pages 1-45

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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APR1400 SUBCOMMITTEE

+ + + + +

THURSDAY

SEPTEMBER 22, 2016

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ROCKVILLE, MARYLAND

+ + + + +

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Ronald G. Ballinger, Chairman, presiding.

COMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

CHARLES H. BROWN, JR., Member

JOSE A. MARCH-LEUBA, Member

DANA A. POWERS, Member

JOY REMPE, Member

PETER C. RICCARDELLA, Member

JOHN W. STETKAR, Chairman

MATTHEW W. SUNSERI, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER L. BROWN

ALSO PRESENT:

TONY AHN, KHNP

MAITRI BANNERJEE, NRR*

JOHN BUDZYNSKI, NRO

BOB CALDWELL, NRO

SEOG NAM CHOI, KHNP

WOOCHONG CHON, KEPCO NF

JEFF CIOCCO, NRO

TIM DRZEWIECKI, NRO

SONG DUK BIN, KEPCO E&C

GORDON HALL, Westinghouse

JOHN HONCHARICK, NRO

JASON HUANG, NRO

GYUN DO JEONG, KHNP

REBECCA KARAS, NRO

JONG SOO KIM, KHNP

JUNGHO KIM, KHNP

MINSEOK KIM, KHNP

YUN HO KIM, KHNP

LIM KWANGIL, KEPCO E&C

SUNGUK KWON, KHNP

HIEU LE, NRO

CHANG-YANG LI, NRO

GREG MAKAR, NRO

RYAN NOLAN, NRO

JIYONG OH, KHNP

ERIC REICHELT, NRO

JAMES ROSS, KHNP/AECOM

ROB SISK, Westinghouse

TAE SUN RO, KEPCO E&C

ANGELO STUBBS, NRO

JEONG KWAN SUCH, KHNP

YOUNG SIL SUL, KEPCO E&C

JESSICA UMANA, NRO

DAN WIDREVITZ, NRO

SEOK JEONG YUNE, KEPCO E&C

* Present via telephone

A G E N D A

P R O C E E D I N G S

8:35 a.m.

CHAIRMAN BALLINGER: This meeting will come to order. This is a meeting of the -- a renewed meeting of the APR1400 Committee of Advisory Committee of Reactor Safeguards. I'm Ron Ballinger, Chairman of the Committee. Members in attendance are Joy Rempe, Charlie Brown, Jose March-Leuba, John Stetkar, Matt Sunseri, Dana Powers and Peter Riccardella.

The purpose of today's meeting is an extension of yesterday's meeting to receive briefings from KHNP regarding -- and the staff regarding the design certification for the APR1400. The rules for today's meeting again were published in the ***Federal Register*** on September 7, 2016. The meeting was announced as an open and closed meeting, so if we have questions that turn out to be proprietary, you need to let us know so we can close the meeting.

A transcript is being kept and made available in the ***Federal Register***, as stated in the ***Federal Register*** notice. We would request that people shut off or disable chicken sounds and stuff like that various cell phones and other items that

we've heard already this morning.

Participants should first identify themselves when they have comments, and speak with sufficient clarity and volume so that they can be readily heard. We have a bridge line open I think again, and before we get started, I want to make some clarifications of yesterday.

This is a subcommittee meeting, which means that questions from members are questions from the subcommittee members themselves. They're not questions from the ACRS. The ACRS communicates through its letters only.

So while we have kept a list of questions and we've gotten asked questions where the applicant or the staff may say we'll get back to you because we need to get an answer, that is not an RAI all right, definitely.

Not an RAI, and so we'll adjudicate those questions amongst the staff and things like that. So that's to be very, very clear about that and I think that's clear under Member Stetkar wants to say something in addition.

MEMBER STETKAR: No, that's pretty clear.

CHAIRMAN BALLINGER: So with that, does

-- you want to say something Bill or Jeff?

MR. CIOCCO: Jeff Ciocco. I'm the lead project manager for the APR1400 design certification project. Thank you for having us back this morning. We look forward to giving you our staff presentation on Chapter 5. We have everybody assembled up here at the table. Over here, we had good representation by our technical staff in the audience, as well as our technical branch chiefs, and we have a division director with us as well today from the Division of Engineering, Bob Caldwell.

So we look forward to giving you our presentation, any questions that you have, thank you.

CHAIRMAN BALLINGER: And so who's up? Jessica.

MS. UMANA: Okay. Good morning. I'm Jessica Umana. I'm the Chapter 5 Project Manager for APR1400. Today, you will hear the staff's presentation to ACRS on reactor, I'm sorry, reactor coolant system and related systems. Specifically, the staff will cover technical topics and open items that came up fast in our review, which was captured in the safety evaluation which you received some

time prior to this meeting.

I'd like to take this opportunity today to just present the presenters. We have John Budzynski, Dan Widrevitz, John Honcharik and Greg Makar. Let me move on through the slides. While you see the presenters up here, I do want to take a moment to appreciate this item. The next is the entire staff that worked on Chapter 5.

On this slide, we have it over on this one and I believe the next two slides we an overview of the SRP sections that were reviewed and the number of questions that were generated from each section with open items. There is one correction I'd like to make, and that's on this slide.

This one shows that we have one open item in 5.4, but that open item is actually in Section 5.4.1.2. The confusion was on my part based on the numbering scheme that was produced when the RAI went out. But I do want to make that clarification. It's not in 5.4. It's 5.4.1.2.

I believe here we have a summary. Overall, 77 questions were generated in Chapter 5, and we have 13 open items. I think maybe one or two may have been brought to resolution since this presentation was prepared by the -- I'll let the

staff cover that.

So as you can see here, go back a few slides. A number of questions were generated in 5.4.1.1, which is the Reactor Coolant Pump Flywheel, and then we had a few questions come from 5.2.3. So now we'll turn it over to John Budzynski. He's going to be presenting in 5.2.2, which is Overpressure Protection, 5.4.1.2, Reactor Coolant Pumps, 5.4.7, Shutdown Cooling, 5.4.12, Reactor Coolant System High-Point Vents.

MR. BUDZYNSKI: Okay. The first section we're going to hear is Section 5.2.2, Overpressure Protection System. The main primary reason for the overprotection is to protect the reactor cooling system, connecting systems and secondary sides of steam generators.

The reactor cooling of the pressure boundary is consistent with four POSRVs, two shutdown cooling relief valves and 20 main steam safety valves, five per steam line and the shutdown cooling suction line relief valve provides sufficient pressure, relief capacity to indicate the most, the main low pressure overprotection event.

Let's go on to the next slide. In the there open items, two have been since closed. The

only remaining open item is the second one, 5.2.2.7 and we requested additional information on how they determine the methodology in what computer codes they used, and then put parameters in the assumptions being made in the analysis for a limiting LTOP event.

MEMBER STETKAR: John, I had a question about that, juts for my edification. When I read through that section of SER, I know you're concerned about energy input from the secondary to the primary system. Is that what --

MR. BUDZYNSKI: Yes.

MEMBER STETKAR: It's noted. It says "The applicant did provide information on the secondary to primary heat transfer using the secondary temperature 230 degrees Fahrenheit, 110 degrees C greater than the RCS, which the staff concludes is substantially more conservative than the 100 degrees Fahrenheit, 37 degrees C difference allowed by the technical specifications, and is therefore acceptably conservative."

How come we have a situation where the secondary temperature is at 230 degrees Fahrenheit higher than the primary temperature under conditions when LTOP is required, and I don't -- because

yesterday I couldn't ascertain whether or not the actual enable temperature was proprietary. So we all know what it is but --

MR. BUDZYNSKI: I would have to say that the gentleman that wrote this RAI is not present, and he is the gentleman that's familiar with this RAI. I would have to go back and discuss it with him --

MEMBER STETKAR: Because that -- see, that's a really big delta T --

MR. BUDZYNSKI: Yes, it is.

MEMBER STETKAR: And I'm not a thermal hydraulics guy, but I sort of know where water boils under what different pressures and how heat transfer normally works.

So it's really hard for me -- when you say it's acceptably conservative, if it's -- if it's absurd, that isn't what we ought to be doing either. So if you could help me with that somehow, you know, just put it on -- it's on the transcript and as Ron said it's just a question.

MR. BUDZYNSKI: Okay.

MS. KARAS: This is Becky Karas.

MEMBER STETKAR: It's only -- I'd hate to see applicants doing analyses with just

artificial conditions, and then somebody saying well that's acceptably conservative.

MS. KARAS: This is Becky Karas, chief of Reactor Systems. I can try and get the reviewer down here.

MEMBER STETKAR: Oh, it would help if it's a quick answer, you know. If it isn't, then we're going to be meeting on these things.

MS. KARAS: Okay, okay, all right.

MEMBER STETKAR: It isn't -- it doesn't necessarily have to be answered, you know, immediately if the cognizant person isn't here.

MS. KARAS: Yeah. I can try and get him down here during this meeting, okay.

MEMBER STETKAR: Thank you.

MS. KARAS: Okay.

MR. BUDZYNSKI: The next section 5.4.1.2, Reactor Coolant Pumps. The pump design is -- the reactor coolant pump is a vertical shaft, single stage centrifugal pump. Loss of cooling water event is the most serious one. CCW flows to the pump motor bearing old cores, motor air coolers, high pressure seal water coolers pump and the CVCS continues to provide sealed water injection flow to the RCP pump by seals.

Pump and motor bearings designed to withstand the loss of CCW for 30 minutes. CCW, that's Division 1, supports charging pump new flow heat exchanger and CVCS and all coolers under RCPS. The seal design is improved to have low seal leakage rates during a station blackout event.

Two open items that have since been closed and resolved so we won't discuss it here.

MEMBER STETKAR: I had, John, a question on the overspeed for the flywheel, and I hit you with it because I beat up the applicant enough yesterday. When I read their section of the DCD, they're very, very, very careful in the DCD to say that the highest anticipated overspeed is predicted for a loss of coolant accident, with the largest break size remaining after the application of leak before break.

That break size is apparently four inches, okay. There's a lot of much bigger piping connections to the reactor coolant system than four inches. How do I treat -- I mean does the staff accept leak before break arguments in terms of LOCA sizes for this type of analysis? What happens if I get like a 12 inch break? Does the flywheel come apart? That's my concern.

(Off mic comments.)

MR. HONCHARIK: I guess this is John Honcharik. So I guess the question is whether or not that four inch line is the limiting line?

MEMBER STETKAR: They claim it's the limiting line after they apply all of their leak before break stuff. In other words, whatever -- I haven't read Chapters 3.6.3, nor am I a specialist in leak before break analyses because that's always mystified me. But apparently after you do your leak before analyses, they conclude that the largest break equivalent size that they can get after all of that is four inches, and they did their overspeed analysis based on a four inch break is my understanding.

It said based on that four inch break, I won't exceed my design overspeed of the pump, and my question is well, you know, the loops themselves are really big and I don't want to get into the double ambiguity and sheer of the loop, and there's pressurize the surge line it's 12 inches.

The direct vessel injection lines are like eight or eight and a half inches. Couldn't find the charging and letdown lines sizes, but they're probably, you know, in the four inch or so

range. What happens if they get a bigger break, you know? Essentially you've accepted their overspeed analysis as being acceptable.

What I don't know is if they have a bigger break, can they get a larger overspeed and have the flywheel come apart?

MR. HONCHARIK: Yeah, because I looked at the flywheel and I know, I mean I know they -- I think they -- I'm sure they did leak before break as you stated, and basically there, instead of having a break you'll have a leak. So more than likely those, what they're saying then is that those larger pipe sizes more than likely will have a leak, not a break.

So some of the limiting ones would be the other ones, and I think the largest that they specified was four inches. But I'd have to consult with someone for the leak before break.

MEMBER STETKAR: Yeah, and indeed that's what they say they did. But what I'm curious about -- and I don't know how that factors into design basis LOCA analyses. That's a different topic. I'm just saying for the purposes of evaluating the integrity of the flywheel or an overspeed is does the staff typically accept those types of leak

before break arguments.

I'm not sure that I've seen that before in other applications, but I must admit that I didn't go back and try to do a thorough search of Chapter 5 to see how people did those overspeed analyses.

MR. MITCHELL: Yeah, and this -- if I could interject. This is Matthew Mitchell, Chief of the Materials and Chemical Engineering Branch in NRO. I believe the characterization with respect to the span of systems which on occasion be applied a leak before Break 2 is accurate and that they did eliminate from a leak before break analysis perspective lines down to -- essentially into the forward trench.

I think you will find that in 363 when we get to covering that. From a philosophical standpoint, in terms of the application of leak before break technology, I think the staff should probably take this back and discuss this.

MEMBER STETKAR: I was more -- the reason I asked you rather than KHNP is that they have used the leak before break argument in a few places throughout the DCD, and to me this is, you know, it's partly technical, it's partly

philosophical. Because you've -- this is first place where I could hang my hat on something that the applicant has explicitly said they've accounted for, it's why I'm kind of raising it now in terms of what's the staff's philosophy in accounting for those arguments.

MR. MITCHELL: And I completely understand the question. I'm sure, as you will recollect, leak before break has inherently been tied to general design criterion 4, and the elimination of the dynamic effects of pipe rupture.

MEMBER STETKAR: Exactly.

MR. MITCHELL: And that has since the initiation of the leak before break concept been a rather well-defined regulatory box in terms of the type of effects, jet impingement, pipe whip, that could be essentially excluded from the design basis based upon the demonstration of leak before break.

MEMBER STETKAR: Yes.

MR. MITCHELL: Having done leak before break analyses myself for many, many years, back when I was in the other office and then my branch deals with LBB now, I will say that I'm not familiar with a prior application that I can recall where it would have been associated with flywheel overspeed.

I'd need to go back and research that a bit more to see whether or not that was ever established by precedent. But that would be within the scope of dynamic effects of pipe rupture.

MEMBER STETKAR: Yeah. I didn't, as I said, I didn't try to go back to the other DCDs that I have and track it, only because I ran out of time. But I didn't recall hearing it and I've sat in on several of the design certifications over the last seven-eight years or so, and I didn't recall hearing that type of argument before.

MR. MITCHELL: And I think I would agree that we probably want to go back and discuss it amongst ourselves a little bit more because I, likewise I'm not sure whether I've run across that.

MEMBER STETKAR: Yeah. Okay, thank you.

MS. KARAS: Dr. Stetkar, at whatever point you want to return to your prior question, we have the reviewer here for that.

MEMBER STETKAR: I'm pretty thrilled right now before I forget what it was.

MS. KARAS: If you could possibly restate your question, it would be helpful.

MEMBER STETKAR: Oh gee. Well, now we're really pressing. Let me go back to where the

heck it was.

CHAIRMAN BALLINGER: Oh, the temperature on the --

MEMBER STETKAR: No. In all seriousness, the concern as I understand it for the energy and for the low temperature overpressure protection analysis was concerning with heat transfer from the secondary side to the primary side, energy transfer from the secondary side -- there you are -- to the primary side.

In the SER, there's a statement that said you're still tracking that as an open item. We know that. So the applicant did provide information on the secondary to primary heat transfer using the secondary temperature of 230 degrees Fahrenheit, 110 degrees C greater than the RCS, which the staff concludes is substantially more conservative than the 100 degrees Fahrenheit, 37 degrees C difference allowed by the technical specifications, and is therefore acceptable and conservative.

My question was for your benefit, because you may not have been here, I don't know whether the LTOP enable and disable temperatures are proprietary information, so I didn't cite those temperatures. We all know what they are. But given

the enable temperature, is it feasible at all to have a secondary temperature 230 degrees higher than that temperature? In other words, how does thermal dynamics work in this situation?

MR. DRZEWIECKI: So there's another event, I'm sorry. Is this in?

MEMBER STETKAR: Well, I think so, yeah.

MR. DRZEWIECKI: It's on, okay. I'm Tim Drzewiecki. I'm from the Systems Branch. We also looked at this in an event in Chapter 15 as well, in which we were looking at some of the overpressure events there. As far as that specific answer to your question, I don't it my fingertips. I can follow up on that.

It was addressed in an RAI because actually the event that I'm thinking of is a startup. It's a startup of an RCP in a random loop. So we asked an RAI on that, trying to get the basis for where those values are from. I can follow up with that RAI value there, but it's addressed as well in Chapter 15. I forget the event number.

However, as far as the tracking of this RAI and why it's an open item, it is that we were not clear on how the calculation was actually done, some of the other inputs and things like that.

That's why it's being tracked as an open item.

MEMBER STETKAR: My question when I stumbled across the actual temperature was, you know, I understand having confidence that you, that the staff must understand how the applicant did their analyses. That's clear. There's no question about that, and it's also clear that the analysis should be appropriately conservative.

But it shouldn't necessarily be unrealistic in the sense of you can't achieve these conditions. In other words, just saying that, you know, I could pick a 1,000 degree temperature difference and that would be even more conservative, but we all know that's physically impossible.

So I was just questioning, you know, what basis. Are there conditions where you could get that type of delta T in that direction at the time when you need LTOP. It's clear there are other conditions where maybe you could get that type of delta T during a cooldown, while the secondary and the primary system is still intact, but now when you're below the -- as cold as you are when you get the LTOP enable or even the disable temperature.

MR. DRZEWIECKI: Yes, I understand that.

MEMBER STETKAR: Because you're talking

about a few hundred degrees on the secondary side.

MR. DRZEWIECKI: Yeah, yeah. So as far as the details, I'm sorry, I don't have it at my fingertips.

MEMBER STETKAR: No, that's fine.

MR. DRZEWIECKI: But it was definitely addressed in the section that's within 15 on the startup --

MEMBER STETKAR: On the startup of the pump.

MR. DRZEWIECKI: Yeah.

MEMBER STETKAR: Okay, because that again, that might have been. I don't want to speculate. I didn't go over in Chapter 15 and try to draw this. I just found it Chapter 5.

MR. DRZEWIECKI: Yeah, okay.

MEMBER STETKAR: Okay.

MR. DRZEWIECKI: Thank you.

MEMBER STETKAR: Thank you.

MEMBER REMPE: Excuse me while you're still there. I'm a little confused on what it is you're still investigating, because what's in the draft DIC says well, the inputs are conservative and I thought I heard today that it's not the inputs anymore; it's just the methodology you're concerned

about at this point or what is it?

MR. DRZEWIECKI: So as far as the language that the inputs, we found them to be -- conservative and acceptable. I really can't speak to that. I don't know if Gordon wrote that. However, the one thing that I will say is as far as the methods and the actual calculation that was done, that's what we have questions on.

MEMBER REMPE: So this is that it was the code. It starts with an O, OBERP, O-B-E-R-P is the name of the code that was used, right?

MR. DRZEWIECKI: It could be.

MEMBER REMPE: Okay, but that -- okay. So I was just looking at the RAI and the response back, and I was just kind of curious on what was the problem. But I guess we'll hear about it later.

MR. DRZEWIECKI: Okay. Just their response to that was the 5.2.2.7 follow-up, because that would be the one that's still on the evaluation.

MEMBER REMPE: Right.

MR. DRZEWIECKI: Okay.

MEMBER REMPE: But there's two parts to it, and I assume that the hand calculations aren't causing the problems when you're talking about the

methods, because that usually isn't a problem with a hand calculation. But the energy, the second one talks about the method of analysis. But anyway, I just am curious about it. But I guess we'll get an update and we'll hear about it later is the answer to the question at this point.

MR. DRZEWIECKI: Yes.

MEMBER REMPE: Thank you.

MR. BUDZYNSKI: Okay. We're on 5.4.7, Shutdown Cooling. Configuration similar to most current PWR designs. Two independent trains. Shutdown cooling pumps are interchangeable with the containment spray pumps. So this is used during shutdown, refueling, startup and it's used in the range of approximately below 450 psig and 350 degrees Fahrenheit.

When we did the evaluation of gas accumulation, we used Generic Letter 2008, RIS 2013-09 and we evaluated it also against NEI 09-10. We have one open item. We accepted the ITAAC that they wrote, but the only problem we have is that the NEI 09-10 was not referenced in the DCD, and we wanted it to be referenced in the DCD because it -- more or less it's the most current gas management procedure we have, guidance we have.

It takes into consideration void testing and it takes into consideration of a gas accumulation high point sections of the piping configuration, and it's all put together one piece and we wanted that to be referenced. As far as the ITAAC, it was acceptable and we can close that out within -- once we get together with the applicant.

MEMBER REMPE: So I have a question about the section with an RAI that was closed out. It's RAI 384-8100 about natural circulation.

MR. BUDZYNSKI: Yes.

MEMBER REMPE: And the response back from the staff. The reason I'm interested in this is just for future evaluations when we look at accident analysis and natural circulation in a CE plant, and I guess they're basing things on some Palo Verde data and in some startup testing and I just was curious. Are you the person who evaluated that?

MR. BUDZYNSKI: Yes, yes.

MEMBER REMPE: Okay. So could you elaborate a little bit more about why you find that acceptable?

MR. BUDZYNSKI: Okay. What we initially looked at, there was a decay heat ratio of

less than .87 I believe it was, and when we have some hazard they come up with that number. We could find it anywhere. If they said it was less than that, then it's acceptable. There would be no problem in natural flow conditions, okay.

They responded and when they responded they showed that it would demonstrate that at full power, the amount of heat being produced for -- I'm trying to -- I don't have the question in front of me, so I'm trying to think of the response.

MEMBER REMPE: It's okay. A good approximation.

(Simultaneous speaking.)

MR. BUDZYNSKI: --heat produced at full power versus the amount of heat produced at the natural flow conditions, recirculation conditions would be less. So therefore you're not exceeding any type of thermal limits at that condition. Basically, that's what they're saying.

And to me, that made sense, but I don't have my notes in front of me to go into detail on that, yeah.

MEMBER REMPE: Well, I pulled the NUREG from the email that did the calculations for this or the testing and the results, and so I guess we'll be

discussing this a little bit more as we go further into the DCD in other sections.

MR. BUDZYNSKI: Yeah. I can get better information that I have on my desk on that.

MEMBER REMPE: Okay, thank you.

MR. BUDZYNSKI: Okay. RCS high point vents. There's no open items here. This is just a description of the technical topics that were covered during evaluation. Do you want me to go over those, or they're just listed with it.

MEMBER STETKAR: I have a question, but it doesn't pertain to any of the bullets here but it does pertain to the vents so --

MR. BUDZYNSKI: Okay, go ahead then.

MEMBER STETKAR: In the SES, this is -- I need to pretty much understand what you thought about it. In the SER, you start off saying the staff reviewed Section 5.4.12 in accordance with the standard review plan. To ensure the adequacy of the RCGVS, to remove from high points in the RCS non-condensible gases that could hinder natural circulation and core cooling after a design basis event.

Later on in that discussion, it says "To address potential gas accumulation in the steam

generator tubes, the applicant described a procedure to operate the reactor coolant pump for short periods of time to transfer non-condensible gases from the tubes to the other RCS high points for venting, therefore in accordance with 10 CFR 50.34(f)(2)(vi) and 10 CFR 50.46(a). Vents are provided for the RCS high points for the APR1400.

I went back and I read the DCD pretty carefully, because I stumbled over that discussion in the DCD when I first read it, and I later concluded that their discussion of using the reactor coolant pumps was completely unrelated to a natural circulation event, which to me makes a lot of sense because, you know, I kind of have natural circulations when I don't have the reactor coolant pumps.

MR. BUDZYNSKI: Right.

MEMBER STETKAR: So I can't bump the reactor coolant pumps to take non-condensibles out of the steam generators during a natural circulation event. You may want to go back and look carefully at the way the SER is written, because I came away from the SER with the conclusion that natural circulation cooling is fine, and the vents are fine because they can bump the reactor coolant pumps to

take gas out of the U tubes during a natural circulation event.

MR. BUDZYNSKI: Okay. I'll take a look at that.

MEMBER STETKAR: And if that -- if that was your intent, I'm very curious about how one can actually do that. As I've said, when I read the DCD, the way the DCD is organized they're really careful to say that they can open the vents to remove non-condensibles from the pressurizer and in particular from the reactor vessel head I think is the only time that they used the words natural circulation.

They talk about bumping the reactor coolant pumps as a way of degassing the system in another part of the discussion, which is more oriented toward normal operation, normal venting of the system, for example, for cooldown or heat up or things like that. So I'm hoping it's just my reading the SER wrong, and that you're not actually taking credit for that process, to facilitate natural circulation cooling.

MR. BUDZYNSKI: Okay. I'll make that an item to go back and talk to the engineer that did the review.

MEMBER STETKAR: It could be a wording situation in the SER, or if it's actually a technical conclusion, I'd like to understand how they came to that.

MR. BUDZYNSKI: Okay.

MS. UMANA: Okay. We have Darren Widrevitz up to present reactor coolant pressure boundary materials.

MR. WIDREVITZ: So in terms of this slide, I take the open item was an editorial concern that we were looking at a little bit later in the game. I think there really is no technical significance to it. So unless you had questions about it, I'd like to entertain questions about the section in general. Does that make sense?

There was -- the text of the DCD implied a particular -- some welding parameters were sufficient to demonstrate non-sensitization, when in fact it was a coupling with the ACM and A262 A or E, and using these parameters that they felt was sufficient to produce non-sensitized product.

CHAIRMAN BALLINGER: I think we went through that to some extent with the applicant yesterday on this A262 and where it's used and I

think we concluded, at least I did, that Practice A262 A and E are -- would never be used on an actual weld.

They might be used as part of weld qualification as a Member Riccardella mentioned, but not as part of any kind of QA on an actual weld on a pipe, since Practice E requires a U bend sample to be taken out of -- to be taken out in Practice A would require basically inducing IGA in the weld itself.

MS. UMANA: Okay, all right. We'll move on. We have Section 5.2.5 on Reactor Coolant Pressure Boundary Leakage Detection, and I'll be covering this slide. Developing requirements of this area are found in GDC 30 and GDC 20, with the acceptance criteria outlined in Reg Guide 1.45 for GDC 30 and Reg Guide 1.29 and 1.45 for GDC 2 (sic).

The staff conducted its review against SRP Section 5.2.5. I'm sorry, the following review -- the review areas are listed here for you. I'll spare you reading all of those. This one was pretty straightforward. There were three RAIs, most requesting clarification and resolution of some inconsistency, and all were statused as confirmatory items. I'm hoping to see those changes in the next

revision to the DCD.

Now there was one COL information item and the staff found it acceptable, because it's consistent with the Reg Guide which is referenced in the COL information item. Do you have -- if you have any questions, I'll defer to the technical reviewer in the audience.

CHAIRMAN BALLINGER: Maybe this is a time to ask a question which I wanted to ask yesterday, and it's related to this but unrelated to this. I searched for, because I'm sensitized, if you will, to steam generator tube leaks. So I'm always after looking to see what is the allowable, unidentified and identified leakage.

And so that number is bandied about in the DCD and in one place it says .5 gallons per minute is unidentified leakage is allowed, and then ten gallons per minute identified leakage. Is that correct?

MS. UMANA: Chang Li?

CHAIRMAN BALLINGER: Because I've seen other numbers in the document and I'm thinking --

MS. UMANA: I have the technical reviewer here. He'll be able to answer that.

CHAIRMAN BALLINGER: And then 150

gallons per minute through one steam generator.

MEMBER STETKAR: That's a typo.

CHAIRMAN BALLINGER: That's a typo? That has to be a typo, okay. I was going to say that.

MR. LI: I am the reviewer for this sections. What you said is correct for unidentified leakage, the limit specified in the tech spec is .5 gallons per minute. For identified leakage is 10 gallons per minute, and there's an interface between steam generator. In steam generator leakage, I think you quote 150 gallons per minute?

CHAIRMAN BALLINGER: Yes.

MR. BUDZYNSKI: That's per day in the tech specs.

CHAIRMAN BALLINGER: Per day.

MR. BUDZYNSKI: There's a typo in the SER.

MR. LI: The typo, yeah.

CHAIRMAN BALLINGER: Okay, thank you.

MS. UMANA: If there are more questions, we're going to move onto -- I think this is a topic that you enjoy very much, the flywheel integrity and I'm going to hand it over to John, and then he can go through his --

MEMBER STETKAR: I apologize for that. I just somehow associate the flywheel with the reactor coolant pump. I didn't look far enough in your presentation, so sorry.

MR. HONCHARIK: Hi, good morning. I reviewed the reactor coolant pump flywheel integrity, and basically the KHNP provided some areas in accordance with the Reg Guide, and that included material selection and fabrication technique for the flywheel, and any pre-service inspection that will be done and also in-service inspections, and also the overspeed testing per the SRP.

Also they provided the flywheel analysis for the Reg Guide, and that basically looked for the critical speed for the ductile and non-ductile fracture mechanisms, and also it included a fatigue crack growth rates for the flywheel.

Next slide. As everyone stated, there were six open items for this area. The first one was basically they didn't specify what method they're going to determine the fracture toughness of the material in question. So basically we asked that. I think they -- per yesterday, they responded about I think a week ago. So we're still evaluating

that, but I think they mentioned something that they're going to use the direct method basically to do testing for it.

The next item was the operating experience. We have some operating experience for the material that they were using. Mostly it was clarifications or what actual operating experience was applicable to this material and situation. So I think they also provided that.

Next was the -- right now the applicant provided an analysis using one-third ultimate strength. However, the SRP specified that they used a one-third yield strength. So basically we asked a question okay, if you're going to use one-third of the ultimate, provide justification why that's acceptable. If not, use the one-third yield per the SRP.

I think they responded, I think they're probably going to use the one-third yield. They're going to --

MEMBER STETKAR: There wasn't a big difference if I recall, was there?

MR. HONCHARIK: Huh?

MEMBER STETKAR: There wasn't a very big difference between those two values, was there?

MR. HONCHARIK: Probably not, yeah.

MEMBER STETKAR: Yeah.

MR. HONCHARIK: So they're going to do that and I think November they're supposed to provide that revised analysis. The other thing that they did was this flywheel is pretty much similar to all the other ones, except this one has hub, some place for a key. They have that hub that attaches it to the rotor.

So basically we're asking that they also do an analysis of that hub, you know, because if that fails, it could potentially release the flywheel. So I think they responded and they're going to do analysis and I think basically they're basing a lot of it on all that's going to be compression. So that will be coming in November along with the other analysis.

And also any inspections that you're going to be doing on that hub should be the same as the flywheel. So they're also responding to that.

The next was the -- this was just a minor one. Basically, they needed to specify what the acceptance criteria for the NDE methods, so that the acceptance criteria for flaw size, it will be bounded by the analysis. So we're just looking to

make sure that that was consistent and they respond to that. I think they're going to put that in the DCD.

Next slide, and the last one was basically to put in the DCD, because they didn't mention anything about annual inspections of the hub. So we asked them that they would need to put some inspection criteria for the hub. Basically, that was our review for the flywheel. Any questions?

(No response.)

MR. HONCHARIK: Thank you.

MS. UMANA: Okay. We're going to move on to Greg Makar. He's going to present on the steam generator program.

MR. MAKAR: The review in Sections 5.4.2.1 and 5.4.2.2 address a variety of materials and design and inspection topics. Our slide here just refers to the one open item that we had, and that is a -- the accident induced leakage performance criterion in the tech specs, which was the value proposed is an acceptable value, but that's also described in a number of places in the tech spec bases.

We found some potential inconsistencies

and weren't certain that it, that the value that we're using was consistent with the accident analyses in Chapter 15. So after discussing that with them in our radiation protection and accident consequences branch, we found that it was really an administrative issue, that there just needed to be some, some edits in the tech spec bases to --

MEMBER STETKAR: That's what I wanted to make sure. I thought that -- I'm glad you clarified that. So it's simply making sure that the citations of things like .3 GPM and .6 GPM in the tech spec bases align with the values that were used for transients and steam line breaks respectively in the Chapter 15 analyses, right. You're not questioning it all, the 150 gallons per day.

MR. MAKAR: Operationally.

MEMBER STETKAR: Operational limit at all. Okay, thanks. Thank you.

MR. MAKAR: So the applicant has since responded to that and clarified that. So we are -- we'll be closing that open item and tracking it as confirmatory.

MS. UMANA: Okay, and just to top off the presentation I provided.

MEMBER STETKAR: Don't go too far yet.

MS. UMANA: Oh.

MEMBER STETKAR: Chris, no Chris.

MS. UMANA: Oh.

MEMBER STETKAR: I know where he's going. I'm trying to turn him off for a second.

MS. UMANA: This is just a courtesy list of the acronyms used throughout Chapter 5 and some that you probably heard in the presentation, if you are interested in referencing it.

MEMBER STETKAR: As usual, they can't shut me up. They try.

CHAIRMAN BALLINGER: We don't want to.

MEMBER STETKAR: Yes, you do but --

CHAIRMAN BALLINGER: Okay, we do.

MEMBER STETKAR: And I'm okay with that. Yesterday, I noted that the applicant has included this Appendix 5A specifically addressing the inter-system LOCA evaluations, and I had not seen in other design certifications such an extensive evaluation of that topic, and I'd like --

Again, I think that's very, very good. It shows a lot of thoughtfulness. I had a question for the staff, and this is -- I saved it for you because it's I think more pertinent for you.

In their evaluations, in particular of

the shutdown cooling system, the safety injection system and containment spray system, they rely substantially, I don't think entirely but substantially, at least in the DCD, on the fact that they designed those piping systems outside the containment to a pressure of 900 pounds, 900 psig, which they say is 40 percent of the normal reactor coolant system operating pressure of 2,250 psi.

I think that's a good idea, you know. They've in fact enhanced the design of those systems. The reason I have a question for the staff, and that I'd like to have a little discussion about this is that they repeatedly refer to things like this is in accordance with the NRC's position presented in the NRC Letter Reference 2.

And the Letter Reference 2 is the NRC letter preliminary evaluation of the resolution of the inter-system loss of coolant accident issue for the Advanced Boiling Water Reactor design pressure for low pressure systems, Docket No. 52-001. So we've got that on the record.

The staff's SER says well, you looked at that 900 pounds and you concluded that it's consistent with past applicant practices that you found acceptable, and you have reasonable assurance

that at this design pressure, the integrity of the RCS will be maintained.

My question is I haven't seen that argument used in any other design certification, and this seems to be establishing a precedent, such that if I just come in and claim that my shutdown cooling system, residual heat removal system or any other low pressure system at 40 percent of my reactor coolant system operating pressure, I get carte blanche. I can't have an interfacing system LOCA, because that is NRC's position.

And in this particular application, I can personally point to other things that provide additional protection in those systems against interfacing system LOCAs, and therefore draw my own conclusion about the susceptibility of those systems to breaks or the likelihood of their being overpressurized and then failing.

But what I'm asking the staff is is this an NRC regulatory position, because as we are all painfully aware, that once this is written in a safety evaluation report for this applicant, it will become regulatory precedent. I think there's a danger in doing that. I think that use of enhanced piping system design is part of the -- part of the

solution.

I think that judicious locations of pressure relief valves, temperature indications, pressure indications, automatic or manual isolation release flow paths and those types of things are all -- testing. For example, how frequently do you go in and test interstitial spaces, to make sure that they're not overpressurized, that you don't have leakage that you perhaps couldn't monitor.

They're all part of the solution of this, not just simply saying well I've designed my piping to greater than 40 percent of reactor coolant system pressure. So I'd ask you to take that and come back and say is that a formal NRC regulatory position regarding interfacing system LOCAs.

The reason I'm concerned about that is twofold. Number one, from just a regulatory footprint, and number two, I certainly don't want to in the future go fighting battles with people doing risk assessments, saying that we don't need to consider thinking about interfacing system LOCA because the NRC staff has told us that we are absolutely safe because our piping is designed to 40 percent of reactor coolant system pressure. We've faced those battles in the past.

So the other thing I'll bring up is that the -- in research, they're currently working on the Level 3 PRA project, and as part of that project, they've had rather extensive work done on interfacing system LOCAs. I don't know what the current status of that is. We had a briefing some time recently, which might have been several months ago.

As part of that process, there was a report produced out at PNNL, and I'll give you the citation. It's PNNL 24783, Expert Elicitation to Support Interfacing System Loss of Coolant Accident Modeling, and the Level 3 PRA project is not using that report verbatim, but they're using information from that report to inform their treatment of interfacing system LOCAs.

I'm curious whether this old letter report, and I trace back a little bit on the history of the thing with NUREG/CRs, not NUREGs. NUREG/CRs, some of which I can't even find in ADAMS because they're so old, whether that is still consistent with the NRC regulatory position, research position and so forth on susceptibility to interfacing system LOCAs.

So if you can kind of take that kind of

long rambling approach. I'm not necessarily -- I'm not right now, because this is a subcommittee meeting, I personally am not questioning the applicant's conclusions regarding interfacing system LOCA.

But I base my conclusion not only on that 900 pounds. I base it on other features of the system's designs that I've looked at and what they claim they will be doing in terms of monitoring and testing and inspection and so forth.

So and I, you know, it's just I had to bring it up as I did yesterday because neither the applicant nor the staff made any mention in their presentations about, you know, Appendix 5A, about this analysis. Yet in the SER, you've addressed it and the applicant obviously has done quite a bit of work, time working on this.

If we're design certification, they're beyond design basis events. I will admit that. They can be really, really important if not addressed sufficiently in the world of risk assessment, as you're well aware, and now I'm done sir.

CHAIRMAN BALLINGER: Thank you. Other questions from members at this point? We're getting

the phone line open. We should hear the eruption of crackles shortly. While we're doing that, are there any folks in the audience that would like to make a comment or a statement?

(No response.)

CHAIRMAN BALLINGER: Yes, it is. It's open. Is there anybody out there on the line? Can you identify yourself or make yourself known so we can verify that this line is open? Maitri's not even up.

MS. BANNERJEE: Yeah. This is Maitri.

CHAIRMAN BALLINGER: She is up. Thank you. Now is there anybody else out there that would like to make a statement?

(No response.)

CHAIRMAN BALLINGER: Hearing none, I think we're done there. Okay. Now in closing, can we go -- we'll go around the table, start with Pete this time.

MEMBER RICCARDELLA: I have no comments.

CHAIRMAN BALLINGER: Dana?

MEMBER POWERS: (No audible response.)

CHAIRMAN BALLINGER: Matt?

MEMBER SUNSERI: No comments, thanks.

MEMBER STETKAR: Nothing more, other

than to say I was pretty impressed with the staff's review of Chapter 5 in particular. Did a pretty, a pretty doggone thorough review. So good job.

CHAIRMAN BALLINGER: Jose?

MEMBER MARCH-LEUBA: No comment.

CHAIRMAN BALLINGER: Charles, Charlie?

MEMBER BROWN: No more comments.

CHAIRMAN BALLINGER: You're not even here. Oh, there you are over there.

MEMBER BROWN: I am.

CHAIRMAN BALLINGER: Joy, I don't know where her name tag is.

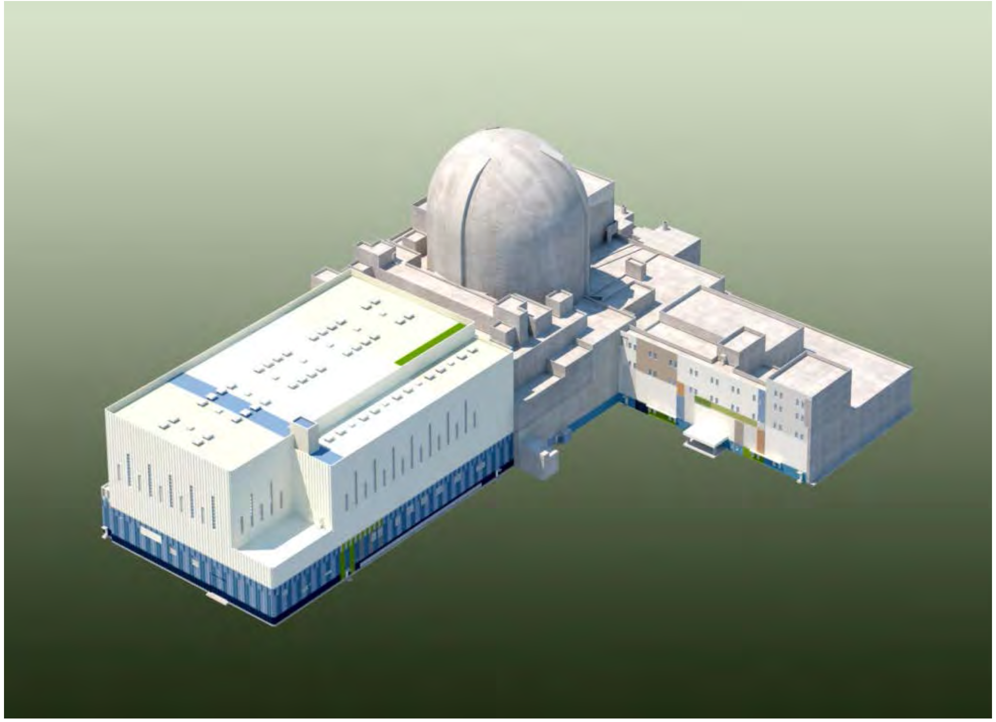
MEMBER REMPE: It's hiding among the electrical cords. But I have no additional comments. Thanks for your efforts and your presentations today. Thank you for your time.

CHAIRMAN BALLINGER: Well, we thank you very much. Again, I would like to express my gratitude for the time and effort and the thoroughness of your presentations, both the applicant and the staff, and with that, we are adjourned.

(Whereupon, the above-entitled matter went off the record at 9:31 a.m.)

NON-PROPRIETARY

Overview of the APR1400 DC Project



KEPCO/KHNP

Sep. 21~22, 2016

KEPCO KHNP

Contents

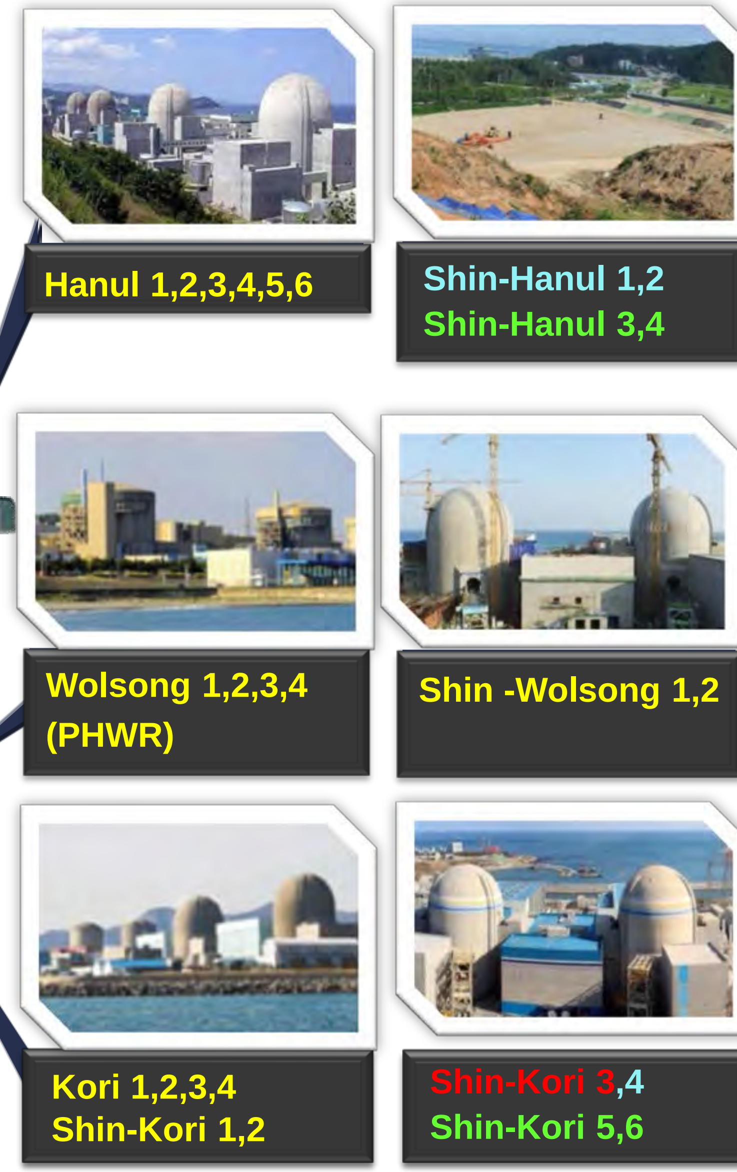
References for design overview of the APR1400

- **List of Presentations to the ACRS in April 20-21, 2016**

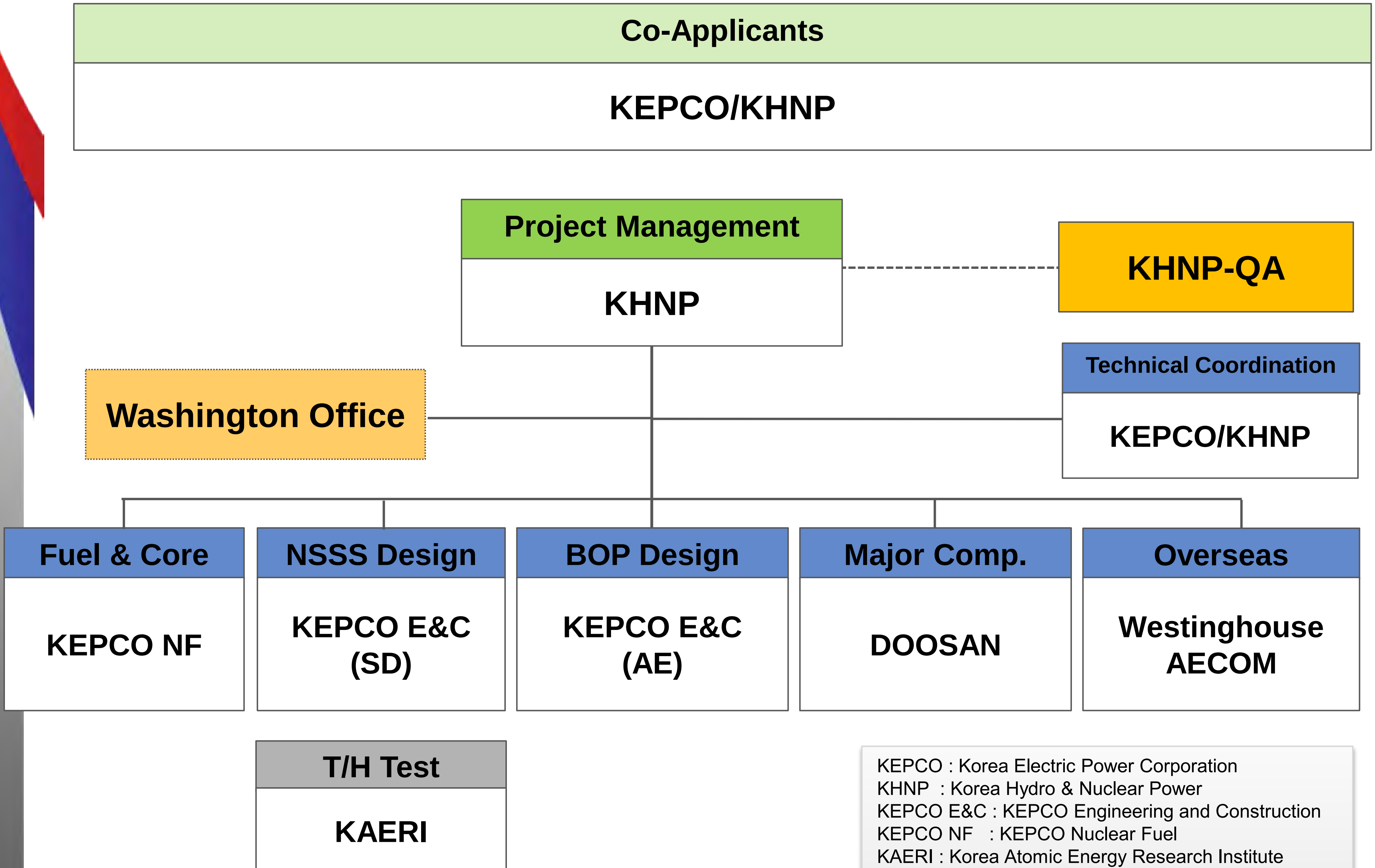
Document No.	Title
APR1400-Z-M-EC-16002-NP	APR1400 system design (Nuclear steam supply system)
APR1400-F-M-EC-16002-NP	APR1400 system design (Fuel design)
APR1400-E-N-EC-16001-NP	APR1400 system design (Containment system)
APR1400-E-C-EC-16001-P/NP	Seismic design analysis
APR1400-Z-M-EC-16001-P/NP	Structural analysis for reactor coolant system
APR1400-E-B-EC-16001-P/NP	Piping design (Piping stress analysis)
APR1400-E-N-EC-16003-P/NP	Piping design (Hazard analysis)
APR1400-E-N-EC-16002-P/NP	Radioactive waste management and radiation protection
APR1400-E-E-EC-16001-P/NP	Electrical system design
APR1400-Z-J-EC-16001-P/NP	Instrumentation & control system
APR1400-K-I-EC-16001-P/NP	Man-machine interface system
APR1400-E-N-EC-16004-P/NP	Plant security design (Aircraft impact analysis)
APR1400-E-N-EC-16005-P/NP	Plant security design (Loss of large area analysis)
APR1400-F-A-EC-16001-P/NP	Safety analysis
APR1400-E-P-EC-16001-P/NP	Probabilistic risk assessment
APR1400-K-X-EC-16002-P/NP	Mitigation strategies for beyond design basis event

Nuclear Power Plants in Korea

In Operation	24 Units	21,716 MW
Commissioning (Reference Plant)	1 Unit	1,400 MW
Under Construction	3 Units	4,200 MW
Planning	6 Units	8,600 MW



Project Organization



KEPCO : Korea Electric Power Corporation
KHNP : Korea Hydro & Nuclear Power
KEPCO E&C : KEPCO Engineering and Construction
KEPCO NF : KEPCO Nuclear Fuel
KAERI : Korea Atomic Energy Research Institute

NON-PROPRIETARY

Project History and Progress

- **[Mar. 2009] Submittal of the Intent of APR1400 DCA to the U.S.NRC**
- **[Apr. 2010~Oct. 2014] Performed total 18 PARMs**
- **[Sep. 2013] First Submittal of the APR1400 DCA to the U.S.NRC**
- **[Dec. 2013] Non-accept of the APR1400 DC application**
- **[Dec. 2014] Resubmittal of the APR1400 DCA to the U.S.NRC**
- **[Mar. 2015] Receive the Docketing letter of APR1400 DC application**
- **[Apr. 2015] Receive the First RAIs[Ch. 2 & 3]**
- **[Jan. 2016] Finished Phase I Review**

21-22. 2016)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NRC NEWS
Office of Public Affairs, Headquarters
Washington, DC 20555-0001
www.nrc.gov ▪ opa.resource@nrc.gov

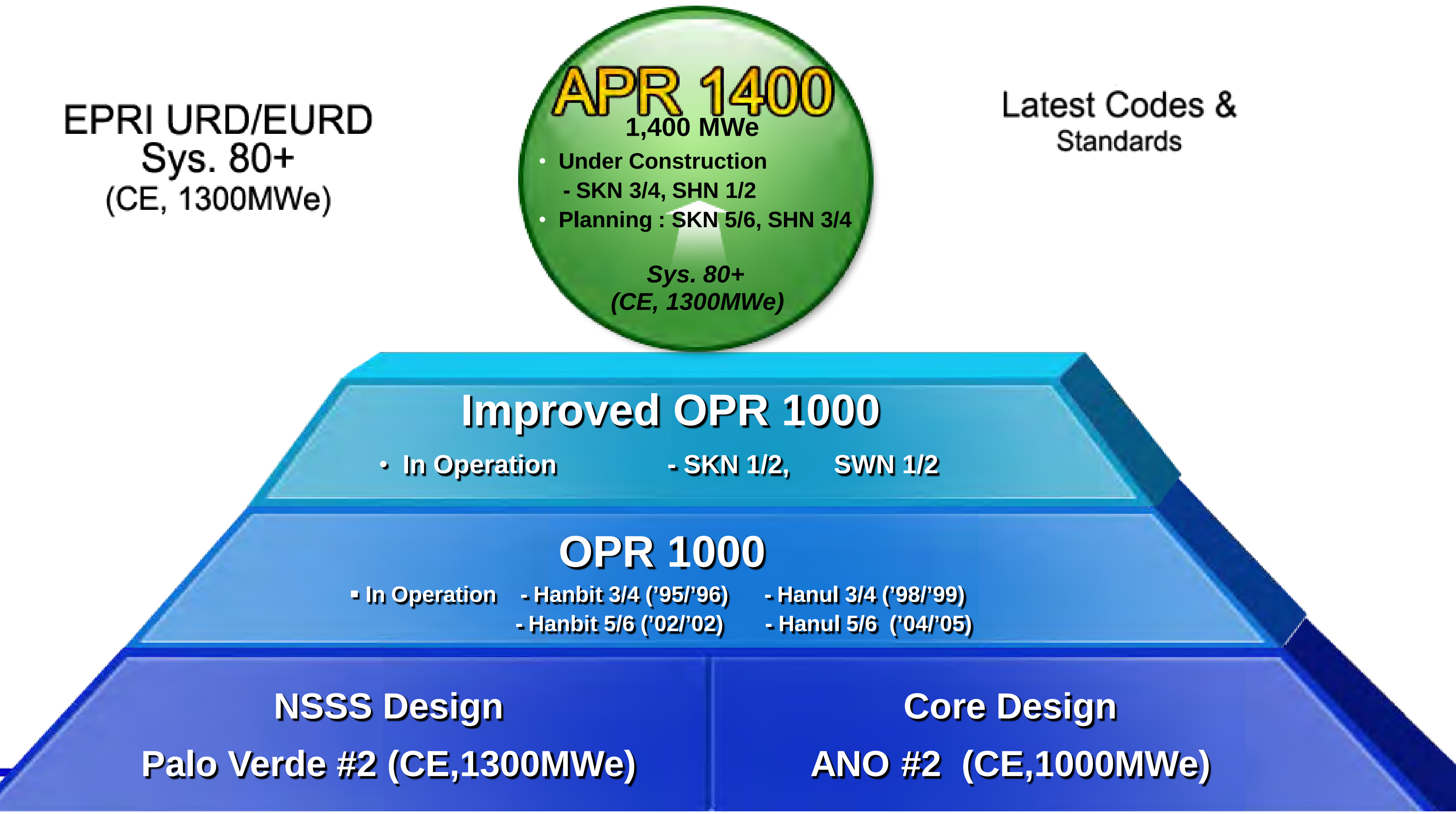
No: 15-012 March 4, 2015
CONTACT: Scott Burnell, 301-415-8200

NRC To Begin Full Certification Review of APR1400 Reactor

The Nuclear Regulatory Commission has docketed for review Korea Electric Power Corp. and Korea Hydro and Nuclear Power's application to certify the APR1400 reactor design for use in the United States.

Development history of APR1400

- Development of Advanced Power Reactor 1400 (1992~2002)
- Licensing agreement with ABB-CE



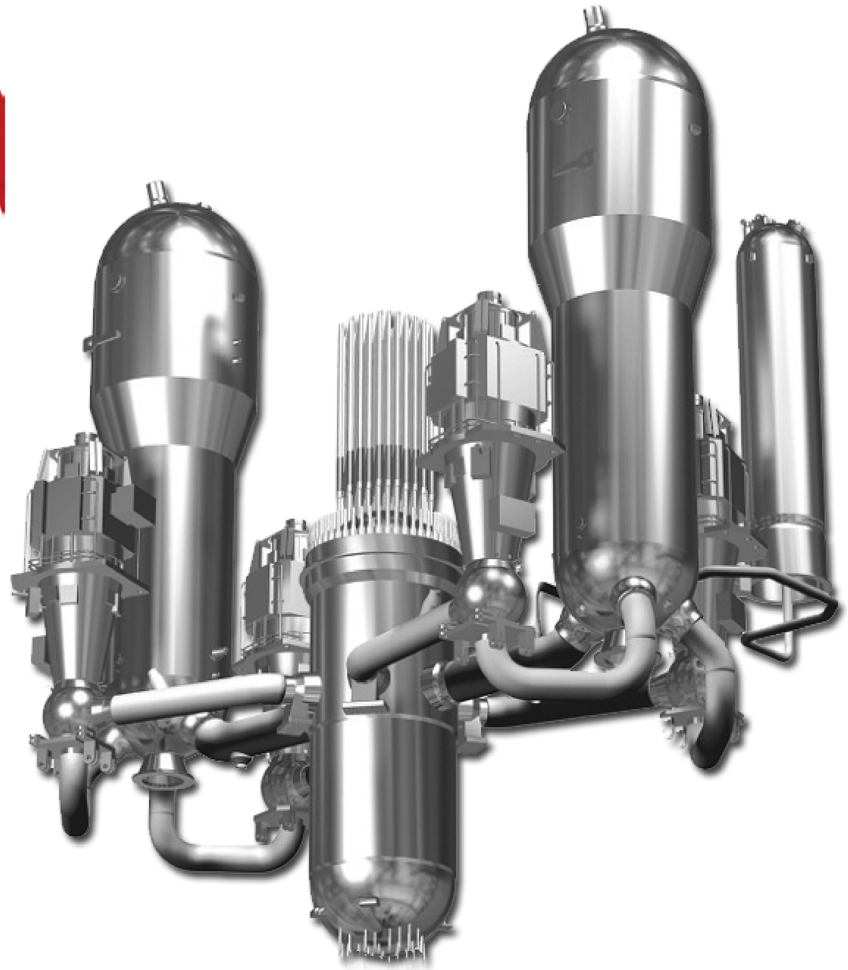
APR1400 Design Features

- **APR1400 referenced Shin Kori units 3&4.**
- **APR1400 is an essentially complete design**
  - Construction completed in Korea (Shin Kori Units 3 & 4)
    - OL for Shin Kori Unit 3 issued on October 2015
    - Criticality reached on December 2015
    - Commercial operation scheduled on October 2016
  - Under-construction in UAE (Barakah Units 1 - 4)

SKN 3&4, Korea

Barakah, UAE

Design Features of the APR1400



- Design Life Time : 60 Years for Class 1 Major Equipment
- Power : 4000MWth / 1400MWe
- Two-Loop : 2 HLs, 2 SGs, 4 RCPs, 4 CLs, 1 Pzr
- Primary Operating condition:
  - Pressure : 2250psia
  - HL/CL Temp. : 615/555 °F
- Secondary Operating condition:
  - Pressure : 1000psia
  - MF/MS Temp. : 450/545 °F
- Pzr Free volume : 2400 ft^3
- SG U-tube : 13102/SG, Alloy 690TT

APR1400 for NRC DC(1/2)

Basic approach of design change for NRC DC

- **Retain reference plant design (SKN 3&4)**
  - To take advantage of proven safety and performance
- **Meet US NRC Regulation Guidance effective on Aug. 2014**
  - Six month before the target docketing date

APR1400 for NRC DC(2/2)

Special Design Considerations for NRC DC

- **Enhance SBO coping capability**
  - Gas turbine generator for AAC source, 16 hr battery, FLEX implementation
- **Improve the tolerance to the beyond design basis**
  - Analysis of aircraft impact by 10CFR50.150
  - Application of LOLA (loss of large area) design requirement
  - Application of physical security requirement
- **Robust design for the design base accidents**
  - GSI-191 for LBLOCA
  - Diverse reactor protection systems for common cause failures
  - Application of FEM model to seismic design

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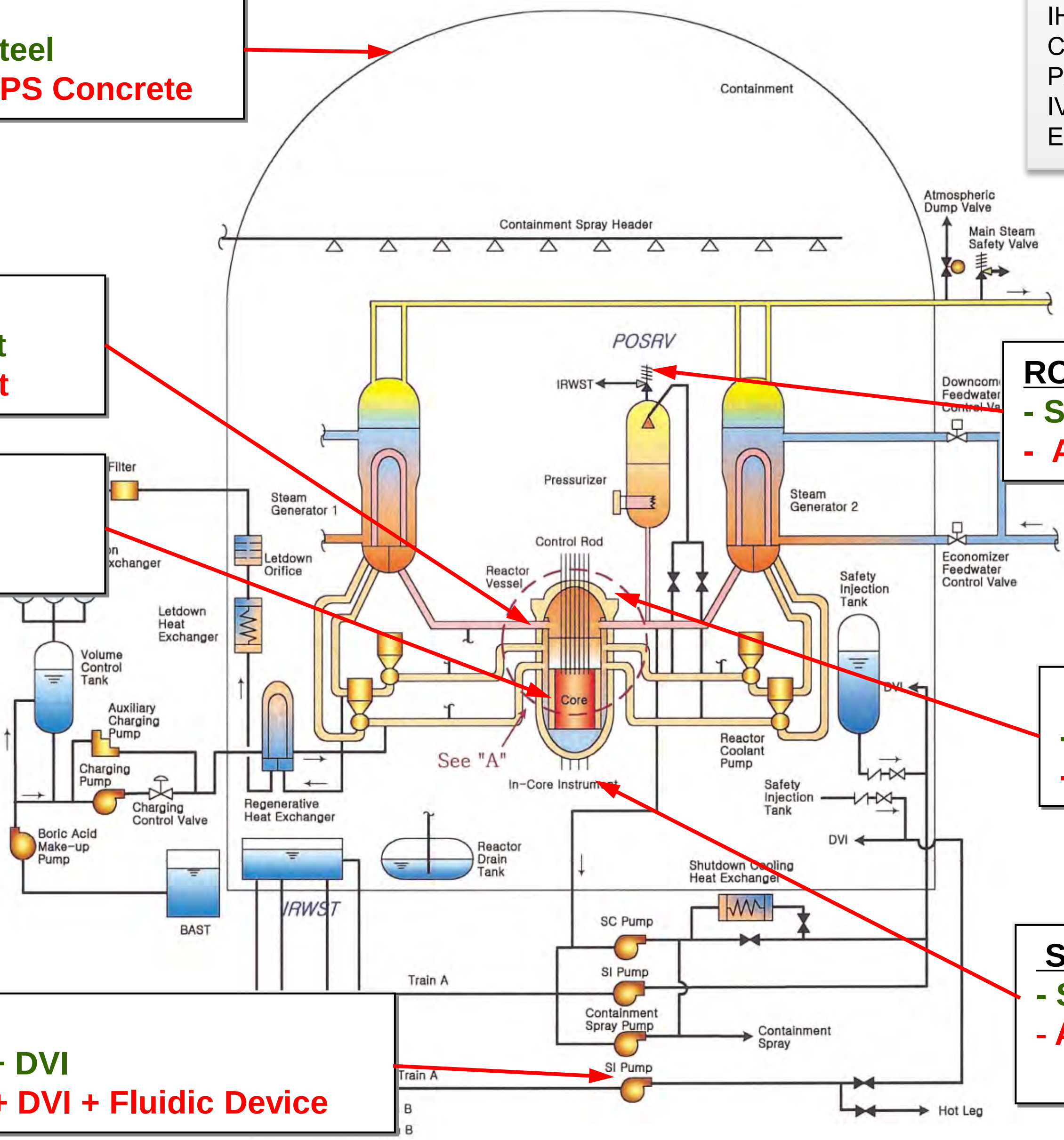
Design Differences between APR1400 and System 80+

SIS: safe injection system
DVI: direct vessel injection
POSRV: pilot operated safety relief valve
IHA: integrated head assembly
CFS: core flooding system
PAR: passive autocatalytic recombiner
IVR: in-vessel retention
ERVC: external reactor vessel cooling

Containment
- System80+ : Spherical Steel
- APR1400 : Cylindrical PS Concrete

Thermal Power
- System80+ : 3,931 MWt
- APR1400 : 4,000 MWt

Hot-leg Temp.
- System80+ : 621F
- APR1400 : 615F



RCS OPP / RD System
- System80+ : 4 PSV + 2 SDS
- APR1400 : 4 POSRV

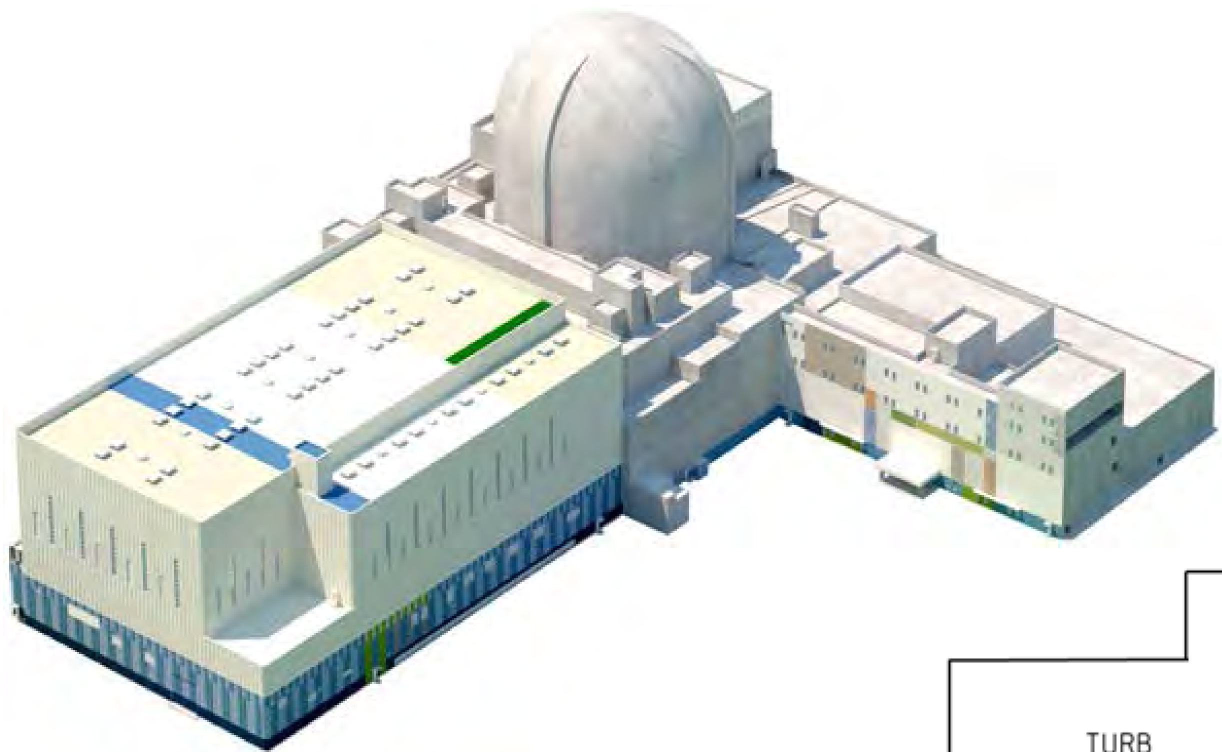
RV Upper Structure
- System80+ : Conventional
- APR1400 : IHA

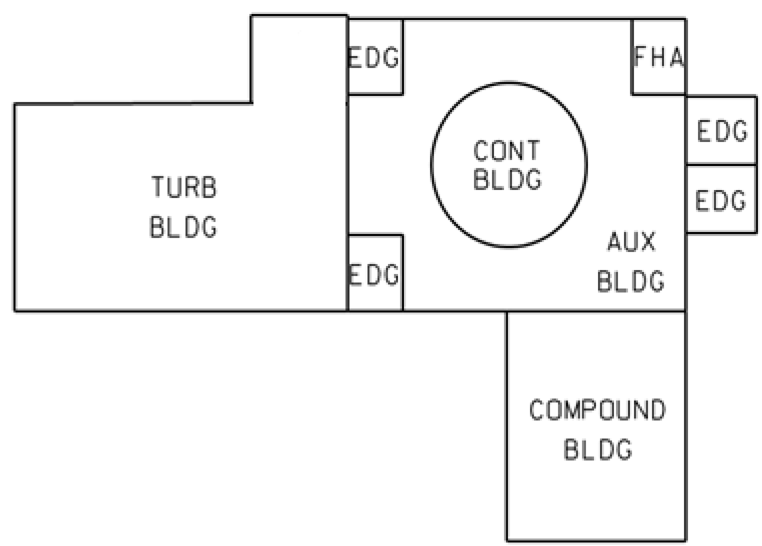
Severe Accident
- System80+ : CFS
- APR1400 : CFS + PAR, IVR/ERVC

Safety Injection System
- System80+ : 4 train SIS + DVI
- APR1400 : 4 train SIS + DVI + Fluidic Device

General Arrangement (1/3)

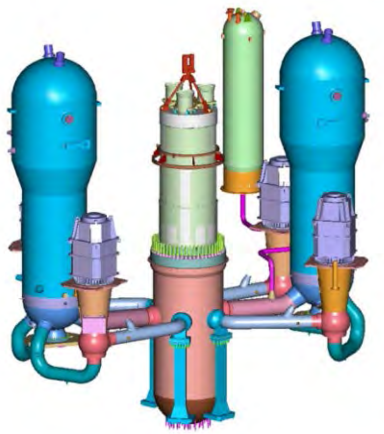
Plant General Arrangement

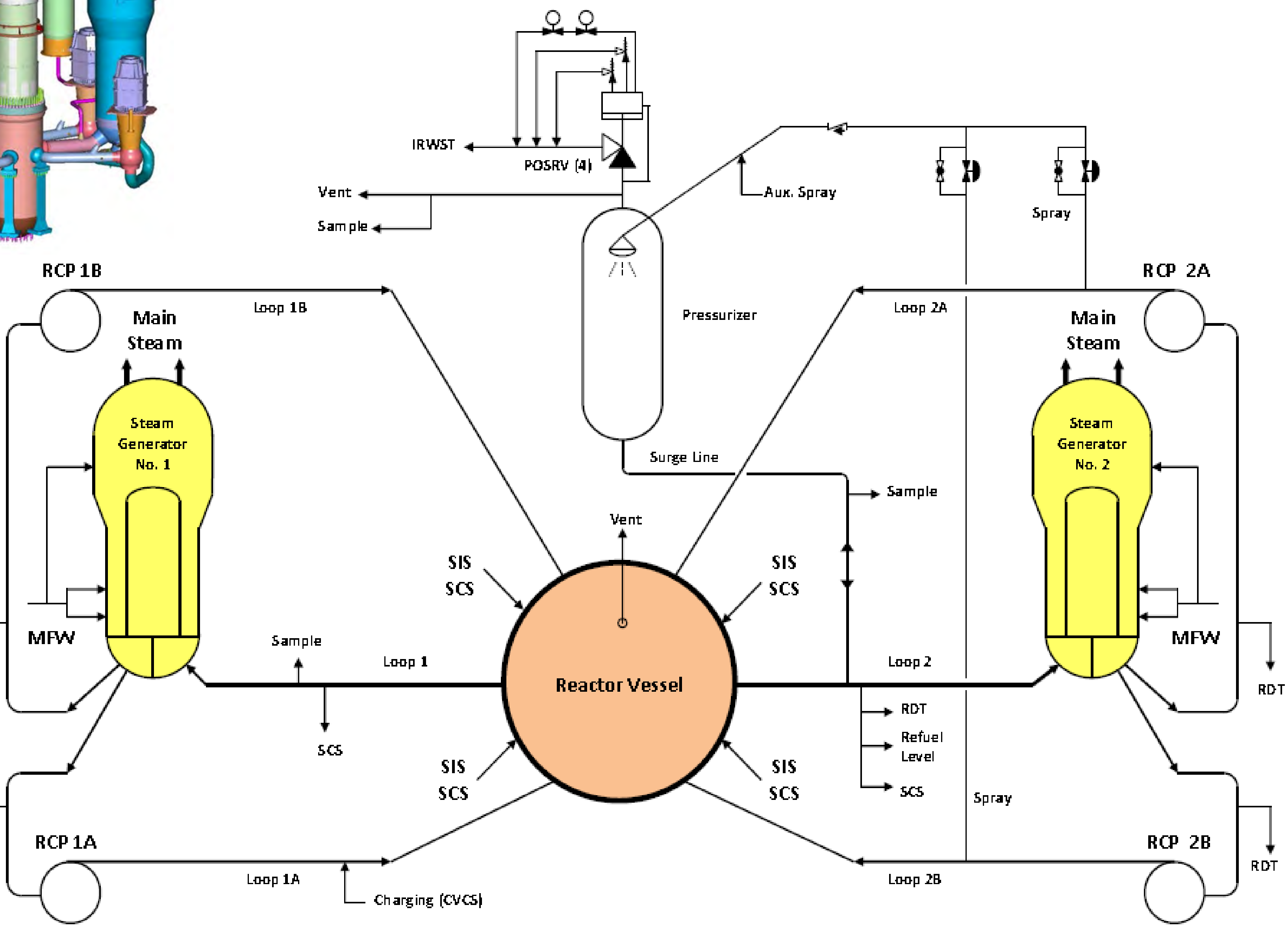




General Arrangement (2/3)

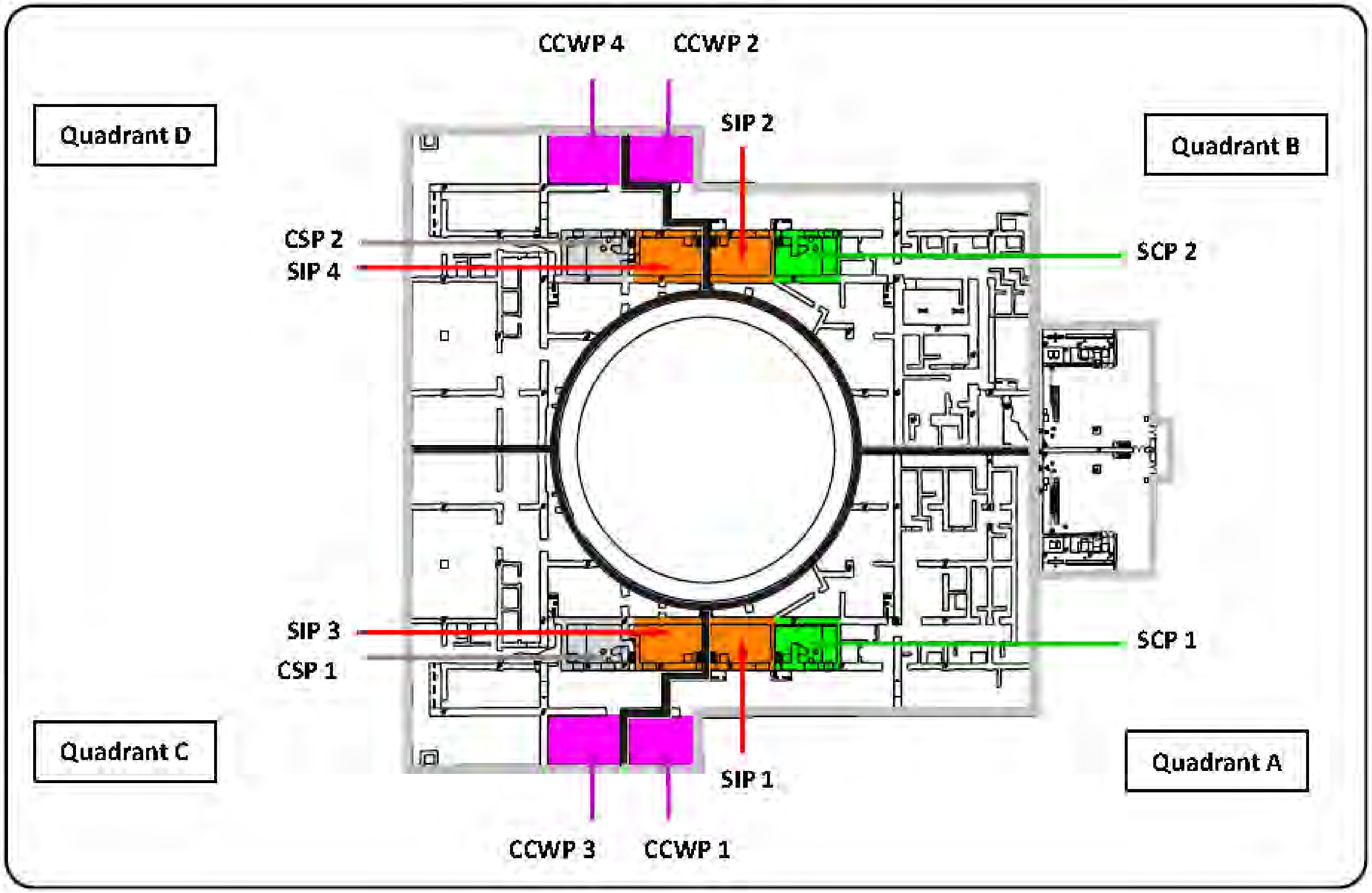
RCS Arrangement Plane





General Arrangement (3/3)

Quadrant Arrangement of Aux. Building



SIP : Safety Injection Pump
SCP : Shutdown Cooling Pump
CSP : Containment Spray Pump
CCWP : Component Cooling Water Pump

Design Review Status

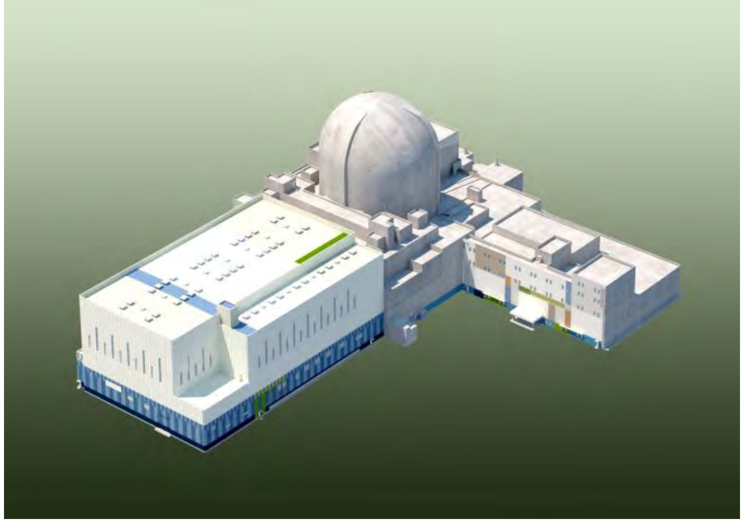
Task	Description	Target Date
Phase Ⅰ	PSER and RAI Completed	Feb. 2016 Jan. 29 2016
Phase Ⅱ	SER with Open Items (Underway)	Nov. 2016
Phase Ⅲ	ACRS Review of SER with Open Items (Underway)	Jun. 2017
Phase Ⅳ	Advanced SER with No Open Items (Underway)	Dec. 2017
Phase Ⅴ	ACRS Review of Advanced SER with No Open Items	Jun. 2018
Phase Ⅵ	Final SER with No Open Items	Sep. 2018

Summary

- **The APR1400 adopted proven technologies from the operation of OPR1000.**
- **The APR1400 used safety analysis codes and methodologies of the certified System 80+.**
- **The APR1400 standard design approval was issued by Korean regulatory authority in 2002.**
  - The first unit of the APR1400, Shin-Kori Unit 3, has been constructed and its commercial operation is scheduled on October 2016.
- **The APR1400 is an essentially complete design.**

Thank you

APR 1400 DCA
Chapter 2: Site Characteristics



KEPCO/KHNP

Sep. 21~22. 2016

Presented by: Kwang-Hoon Koh, Dong-Su Lee

Contents

Overview of Chapter 2

List of Submitted Documents

Document No.	Title	Revision	Type	ADAMS Accession No.
APR1400-K-X-FS-14002 -P & NP	APR1400 Design Control Document Tier 2: Chapter 2 Site Characteristics	0	DCD	ML15006A041
APR1400-K-X-IT-14001 -P & NP	APR1400 Design Control Document Tier 1	0	DCD	ML15006A039

RAI Summary

No. of Questions	No. of Responses	Not Responded	No. of OI
33	33	0	0

Overview of Chapter 2

- **Scope**

This chapter provides the site interface requirements for the APR1400 Design including geological, seismological, hydrological and meteorological characteristics.

The COLA is to confirm site characteristics are bounded, or provide site specific qualification.

Site Characteristics

- The APR1400 is designed on the basis of a set of assumed site-related parameters.
- The parameters were selected to include a range of potential nuclear power plant sites in the United States.
- Table 2.0-1 is a summary identifying specific site parameters for the APR1400 design. (Table provided attachment)

Site Characteristics (cont.)

➢ Subsection Outline

Section	Title	Description
2.1	Geography and Demography	Site Specific Characteristics The COL applicant will provide site-specific Information * on the site location and a description of the site, exclusion authority and control, and population distribution, as stated in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.206
2.2	Nearby Industrial, Transportation, and Military Facilities	Site Specific Characteristics See above

* Complete list of COL Information Items for Chapter 2 is provided at end of presentation

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.3	Meteorology	1. Snow (winter precipitation) a. 100 year snowpack roof load(normal) : 2.873 kPa (60 psf) b. Extreme winter precipitation roof load : 5.985 kPa (125 psf) c. Depth of 48-hr probable maximum winter precipitation (PMWP) : 914.4 mm (36 inch) d. Basis : HMR (NOAA), ASCE/SEI 7, EPRI URD

Site Characteristics (cont.)

- Subsection Outline (cont.)

Section	Title	Description
2.3 (cont'd)	Meteorology	2. Extreme wind a. 50 year 3-second wind gust speed : 64.8 m/s (145 mph), exposure category C b. Importance factor : 1.15 for seismic category I/II structures c. Basis : ASCE/SEI 7 3. Tornado a. Maximum tornado wind speed : 102.8 m/s (230 mph) b. Maximum translational speed : 20.6 m/s (46 mph) c. Maximum rotational speed : 82.2 m/s (184 mph) d. Radius of maximum rotational speed : 45.7 m (150 ft) e. Maximum pressure differential : 8.274 kPa (1.2 psi) f. Rate of pressure drop : 3.447 kPa/s (0.5 psi/s) g. Basis : RG 1.76 4. Hurricane a. Maximum 3-second wind gust speed : 116 m/s (260 mph) b. Basis : RG 1.221

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.3 (cont'd)	Meteorology	5. HVAC outdoor design temperature a. 5% exceedance values : • maximum : 35.0°C (95°F) dry bulb, 25.0°C (77°F) coincident wet bulb • minimum : -20.6°C (-5°F) dry bulb b. 1% exceedance values : • maximum : 37.8°C (100°F) dry bulb, 25.0°C (77°F) coincident wet bulb • minimum : -23.3°C (-10°F) dry bulb c. 0% exceedance values (historical limit excluding peak < 2 hours) : • maximum : 46.1°C (115°F) dry bulb, 26.7°C (80°F) coincident wet bulb • minimum : -40.0°C (-40°F) dry bulb

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.3 (cont'd)	Meteorology	6. Ambient design temperature for cooling tower a. 5% exceedance values for circulating water system • maximum : 26.1°C (79°F) non-coincident wet bulb, • minimum : -20.6°C (-5°F) dry bulb b. 0% exceedance values for essential service water system (historical limit excluding peak < 2 hours) • maximum : 27.2°C (81°F) non-coincident wet bulb • minimum : -40.0°C(-40°F) dry bulb

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.3 (cont'd)	Meteorology	7. Atmospheric dispersion factors a. Purpose • Dose to the public during normal operation • Radiological consequence analyses for DBAs • MCR/TSC habitability analysis b. Regulatory Basis • SRP 2.3.4 – 2.3.5 (Rev.3) • RG 1.23 (Rev.1), 1.111 (Rev.1), 1.145 (Rev.1), 1.194 c. Bounding atmospheric dispersion factors and deposition factors at the offsite locations are presented in Table 2.0-1. d. The 95th percentile on-site atmospheric dispersion factors for the MCR/TSC habitability analysis are presented in Table 2.3-2 through 2.3-12. e. The input variables including post-accident gaseous vent and intake locations used for on-site χ/Qs are shown in Table 2.3-13 and Figure 2.3-1.

NON-PROPRIETARY

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.3 (cont'd)	Meteorology	7. Atmospheric dispersion factors (cont'd) f. Design features • Long-term χ/Q & D/Q : - Used the bounding values of 2.0E-05 s/m^3 for χ/Q and 2.0E-07 m^{-2} for D/Q at EAB to bound most US sites • Short-term χ/Q : - Used 1.0E-03 s/m^3 at EAB based on EPRI-URD • On-site χ/Q : - Selected the most limiting meteorological condition among 6 U.S. NPP sites - Added sufficient margin to envelope most of U.S. sites - ARCON96 code is used

Site Characteristics (cont.)

Postulated Site Parameter Values for APR1400 Compared to Corresponding Site Characteristics and Other Design Site Parameter Values

Site / Design	Site Boundary Annual Average X/Q (s/m³) (No Decay/Undepleted)	Site Boundary Annual Average X/Q (s/m³) (2.26-Day Decay/Undepleted)	Site Boundary Annual Average X/Q (s/m³) (8.00-Day Decay/Depleted)	Site Boundary Annual Average D/Q (1/m²) (Relative Deposition)
APR1400	2.00E-05	1.99E-05	1.84E-05	2.00E-07
	805 m	805 m	805 m	805 m
Fermi (RWB Vent Stack)	1.10E-05	1.10E-05	1.00E-05	4.60E-08
	981 m	981 m	981 m	772 m
South Texas Project	8.10E-06	8.10E-06	7.30E-06	6.40E-08
	1,115 m	1,115 m	1,115 m	1,115 m
V.C. Summer	5.80E-06	5.80E-06	5.30E-06	1.60E-08
	805 m	805 m	805 m	805 m
Vogtle	5.50E-06	5.50E-06	5.00E-06	1.70E-08
	800 m	800 m	800 m	800 m
Clinton ESP	2.04E-06	2.04E-06	1.84E-06	1.46E-08
	1,025 m	1,025 m	1,025 m	1,025 m
Grand Gulf ESP	8.80E-06	-----------	-----------	1.20E-08
	1,368 m	-----------	-----------	933 m
North Anna ESP	3.70E-06	3.70E-06	3.30E-06	1.20E-08
	1,416 m	1,416 m	1,416 m	998m
PSEG Nuclear ESP	1.00E-05	1.00E-05	9.50E-06	4.10E-08
	386 m	386 m	386 m	386 m
AP1000 DCD	<= 2.00E-05	-----------	-----------	-----------
	805	-----------	-----------	-----------

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.4	Hydrologic Engineering	1.Groundwater and Flood elevation a. Groundwater elevation : 0.61m (2 ft) below plant grade b. Maximum level for flood : 0.3 m (1 ft) below plant grade c. Basis : EPRI URD 2. Maximum rainfall precipitation rate (1 mi^2) : a. hourly 492.7 mm (19.4 inch) per hour b. short-term of 157.5 mm (6.2 inch) in 5 minutes c. Basis : HMR (NOAA), EPRI URD

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.5	Geology, Seismology, and Geotechnical Engineering	1. SSE : 0.3g peak ground acceleration 2. CSDRS (Certified Seismic Design Response Spectra) : a. SSE is based on CSDRS. b. Enhanced spectra in high frequency range of RG 1.60 : Figure 2.0-1 and 2.0-2 • Frequency at the PGA is changed from 33 Hz to 50 Hz. • Spectral amplitude at 25 Hz are set to RG 1.60 response spectra at 25 Hz scaled by factor 1.30 . 3. HRHF (Hard Rock High Frequency) RS: a. Peak ground acceleration : 0.46g b. Response spectra : Figure 2.0-3 and 2.0-4

Site Characteristics (cont.)

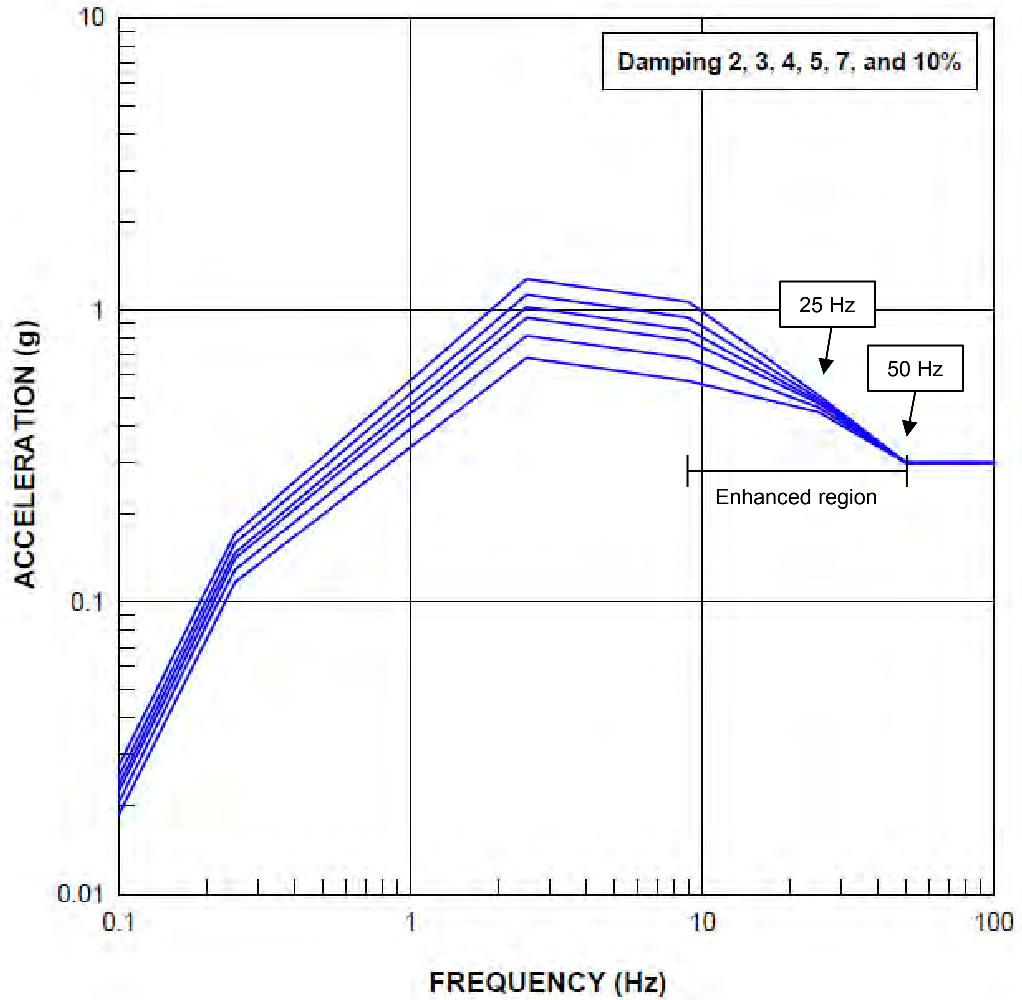


Figure 2.0-1 Horizontal Certified Seismic Design Response Spectra

Site Characteristics (cont.)

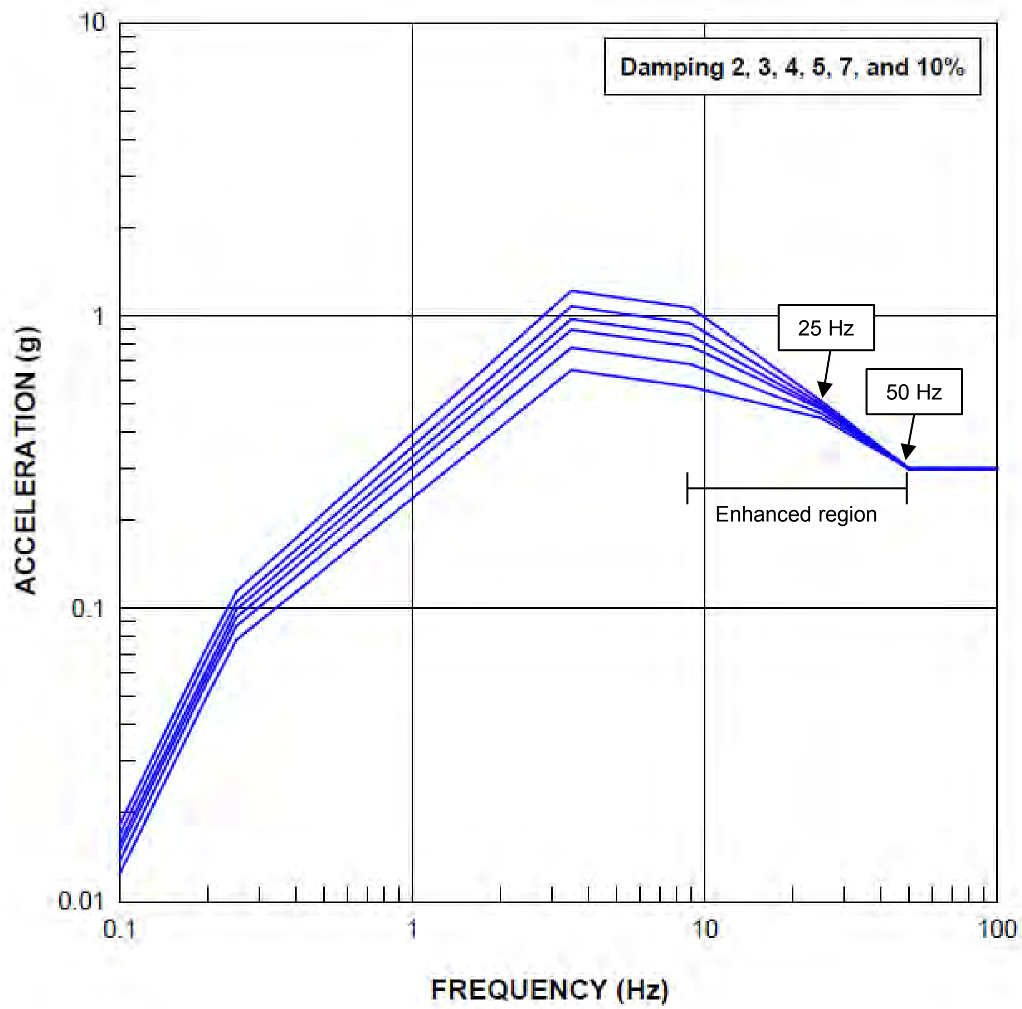


Figure 2.0-2 Vertical Certified Seismic Design Response Spectra

Site Characteristics (cont.)

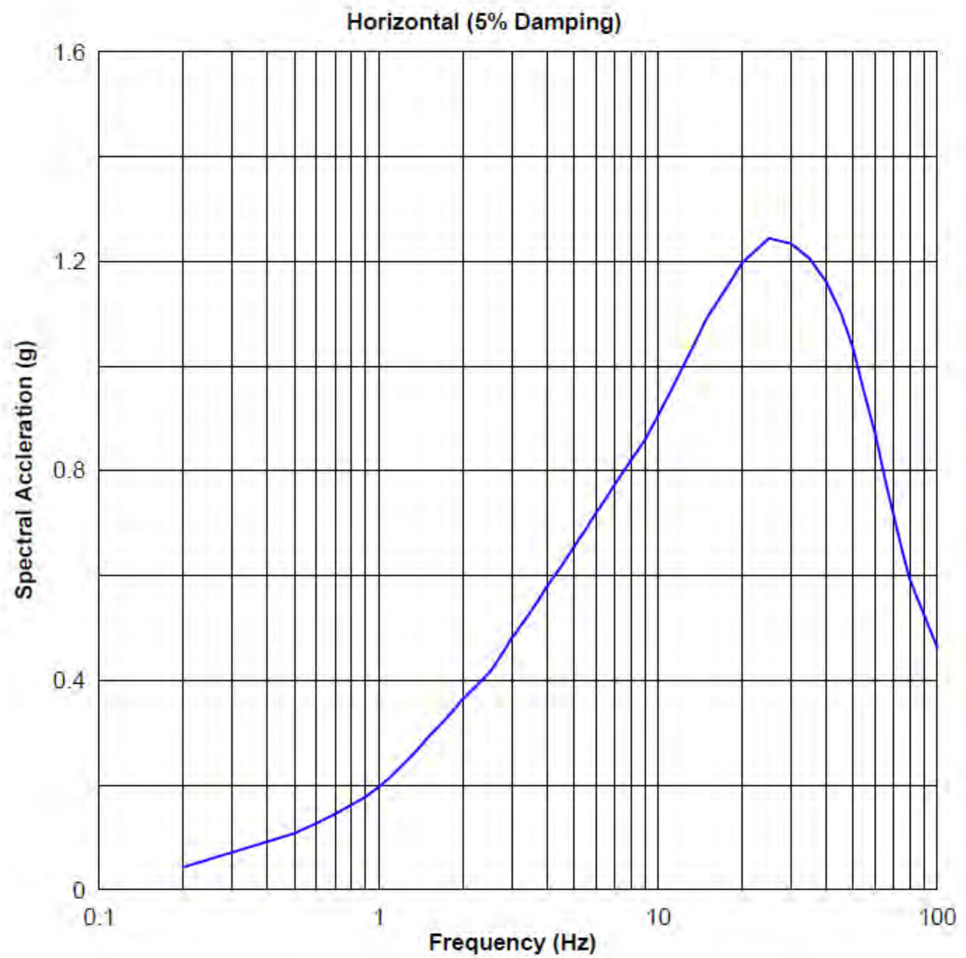


Figure 2.0-3 Horizontal HRHF Response Spectra

Site Characteristics (cont.)

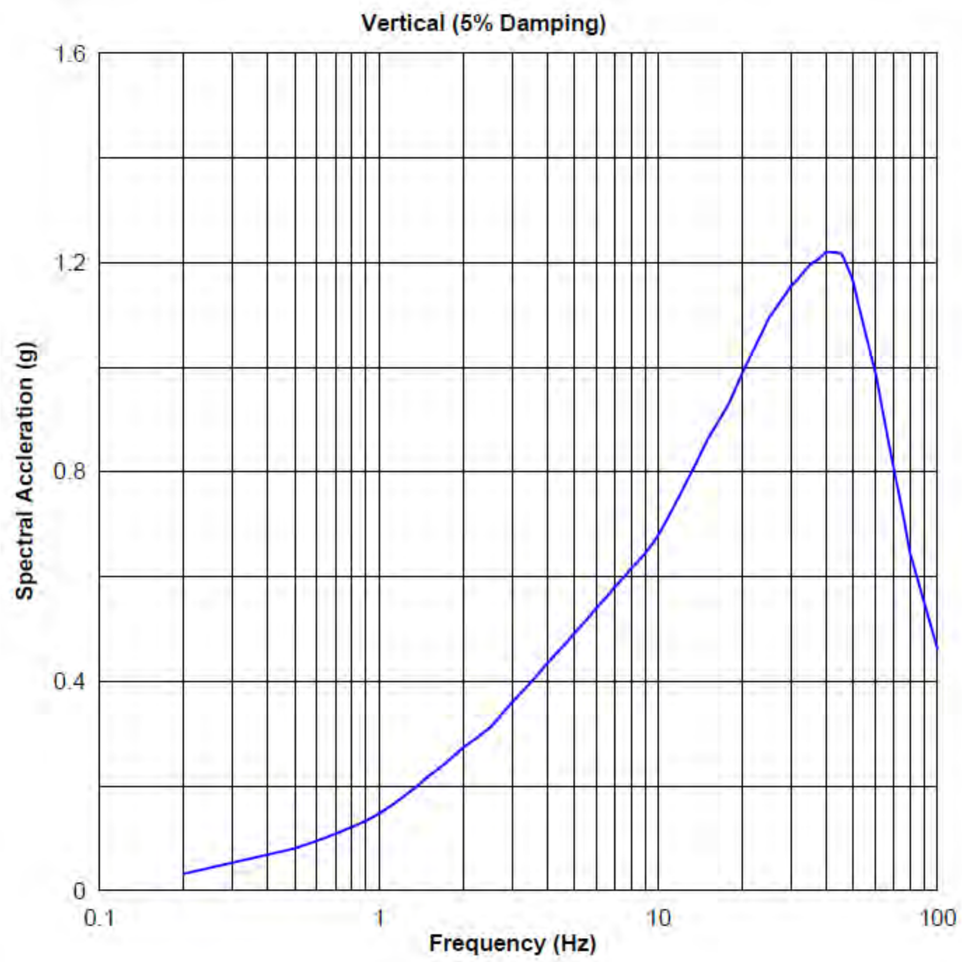


Figure 2.0-4 Vertical HRHF Response Spectra

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.5 (cont'd)	Geology, Seismology, and Geotechnical Engineering	4. Surface tectonic and nontectonic deformation at site : a. The COL applicant will check the deformation potentials and confirm that the effects are within the design basis of the facility. 5. Maximum dip angle for soil uniformity : 20 deg. 6. Subsurface stability: a. Minimum allowable static bearing capacity : 712 kPa (15,000 psf) b. Minimum allowable dynamic bearing capacity, normal conditions plus SSE : 2872.8 kPa (60,000 psf) c. Minimum shear wave velocity at SSE input at ground surface : 1,000 ft/s d. Liquefaction potential : None (for Seismic Category I structures) e. Minimum soil angle of internal friction : 35 degrees below the footprint of the Seismic Category I structures at their excavation depth

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

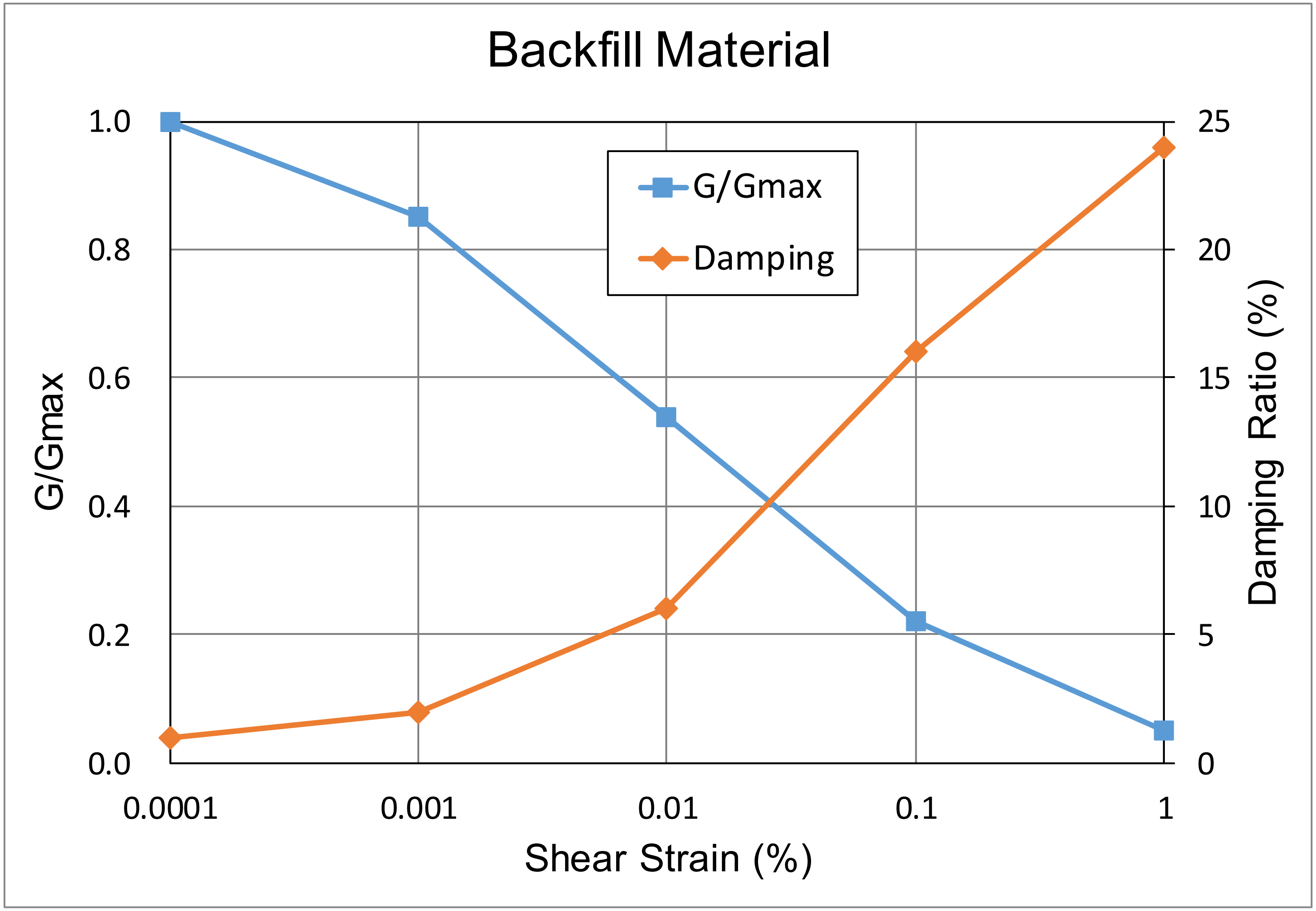
Section	Title	Description
2.5 (cont'd)	Geology, Seismology, and Geotechnical Engineering	7. Allowable settlement a. Maximum allowable differential settlement inside building : 12.7 mmm (0.5 inch) per 15.24 m (50 ft) in any direction for the Seismic Category I structures b. Maximum allowable differential settlement between buildings : 76.2mm (3.0 inch) between NI Common Basemat and EDG Building & DFOT Building, 12.7 mm (0.5 inch) between other adjacent buildings

Site Characteristics (cont.)

➢ Subsection Outline (cont.)

Section	Title	Description
2.5 (cont'd)	Geology, Seismology, and Geotechnical Engineering	8. Backfill material a. Density : 2.2 g/cm^3 (137 pcf) b. Dynamic Poisson's ratio : 0.33 c. G/G_{max} vs shear strain relation d. Damping vs shear strain relation Shear Strain(%) — G/G_{max} — Damping (%) -- 1.0 — 0.05 — 24.0 0.1 — 0.22 — 16.0 0.01 — 0.54 — 6.0 0.001 — 0.85 — 2.0 0.0001 — 1.00 — 1.0

Site Characteristics (cont.)



<G/G_{max} vs Shear Strain & Damping vs Shear Strain for Backfill>

Site Characteristics (cont.)

➢ List of RAIs

RAI No.	Question No.	Description
8012	02.03.01-1&2	Update the parameter table
8012	02.03.01-3	HVAC design temperature in Table 2.0-1
8211	02.03.04-4	MCR and TSC Location
7912	02.03.04-3	Offsite χ/Q Values
CQ*	02.03.04-1	Adding onsite χ/Q into Table 2.0-1
7912	02.03.04-2	Request Met. Data
CQ*	02.03.04-3	Explanation of the Met. Data in Table 2.3-13
8211	02.03.04-5	The source – receptor pairs and The most limiting Met. Data of 6 US sites.
7912	02.03.04-1	Revision of the all calculation without reduction factor of 2
7913	02.03.05-1	Confirmation of the χ/Q and D/Q

* CQ : Clarification Question

Site Characteristics (cont.)

List of RAIs (cont.)

RAI No.	Question No.	Description
8012	02.03.01-4	Extreme winter precipitation roof load
7827	02.05.04-1	Soil Uniformity Requirement
7827	02.05.04-2	Dynamic bearing capacity
7828	02.05.02-1	Applicable NRC Reg. Guides
7829	02.05.03-1	Tectonic surface deformation
7847	02.05.04-3	Minimum soil angle of internal friction
7847	02.05.04-4	Minimum dynamic properties requirements for Structural Fill Granular
7847	02.05.04-5	Lean concrete backfill
7847	02.05.04-6	Minimum dynamic shear modulus
7847	02.05.04-7	Potential for Liquefaction

Site Characteristics (cont.)

➢ List of RAIs (cont.)

RAI No.	Question No.	Description
7847	02.05.04-8	Static and dynamic parameter for EDG and DFOT
7847	02.05.04-9	Static and dynamic bearing demand and capacity
7847	02.05.04-10	Maximum differential settlement
7848	02.05.02-3	Soil profiles
8018	02.05.05-1	Stability of slope

✓ All RAIs were address during the NRC Staff's review and no open items were identified .

Key Review Items

➢ **RAI NO. 8012 Question 02.03.01-3a, NRC provided 8/4/15, KHNP addressed 11/4/15**

- **Description of Issue** : NRC staff noted that almost all the COL and ESP applications previously reviewed identified 0-percent exceedance non-coincident wet-bulb temperatures greater than the corresponding site parameter value (81 ℉) for APR1400.
- **Resolution :** NRC had certified the U.S. Advanced Boiling-Water Reactor design in May 1997 and had recently performed a detailed review of the U.S. Evolutionary Power Reactor (EPR) design, in which plants, 81 ℉ for the non-coincident wet-bulb temperature site parameter was used as part of the US ABWR certified design and as part of the US EPR review. KHNP stated that APR1400 design ambient temperature site parameters and values for all percent exceedance levels are the same as those specified in EPRI URD for Advanced Light Water Reactors. Therefore, the 0 percent exceedance wet bulb temperature of 27.2 °C (81 °F) used in the APR1400 DCD FSAR is reasonable and valid for use today.

Key Review Items (cont.)

➢ **RAI NO. 7912 Question 02.03.04-1 Rev. 2, NRC provided 6/1/15, KHNP addressed 6/1/16**

- **Description of Issue** : Since both the air intakes of the Aux. Building (No. 3 & 4 in the figure) are in the same wind direction window (i.e., within 90°) from the release point 7, the two intakes do not meet the criteria in RG 1.194 for a reduction factor of 2.
- **Resolution** : KHNP updated the χ/Q calculation without credit of the reduction. The resultant MCR/TSC doses for all DBAs are within the acceptance criteria.

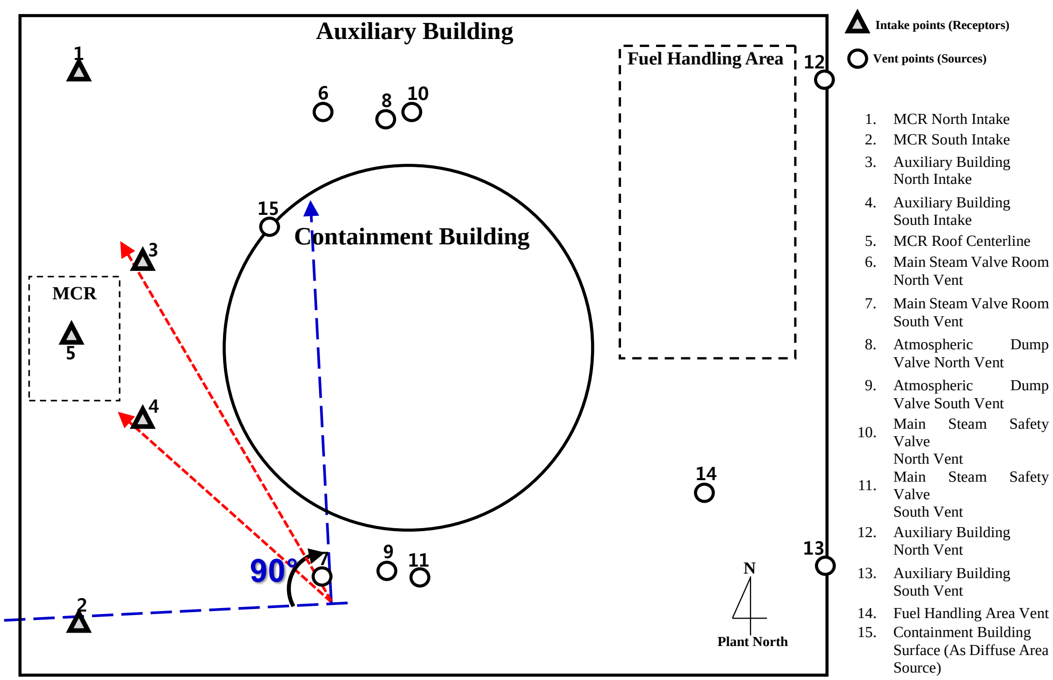


Figure 2.3-1 Locations of Post-accident Gaseous Vents and Intakes

Summary

- Chapter 2 defines site parameters of the APR1400 standard plant and also identifies important site parameters.
- The COLA is to confirm that DCD site parameters envelope site-specific parameters.
- Table 2.0-1 identifies the site parameters for the APR1400 Design.

Acronyms

- KHNP: Korea Hydro & Nuclear Power
- COL / COLA: Combined License / Combined License Applicant
- PMWP: Probable Maximum Winter Precipitation
- HMR: Hydrometeorological Report
- NOAA: National Oceanic and Atmospheric Administration
- ASCE: American Society of Civil engineers
- SEI: Structural Engineering Institute
- HVAC: Heating, Ventilation, and Air Conditioning
- MCR / TSC: Main Control Room / Technical Support Center
- DBA: Design Basis Accident
- SRP: Standard Review Plan
- RG: Regulatory Guide
- EAB: Exclusion Area Boundary
- SSE: Safe Shutdown Earthquake
- PGA: Peak Ground Accelaration
- G/Gmax: Shear Modulus of Soil/Maximum Shear Modulus of Soil

Attachment

➢ Table 2.0-1 Site Parameters (1/4)

Parameter Description	Parameter Value
Maximum Elevation of Groundwater	0.61 m (2 ft) below plant grade (1) in the vicinity of the SSCs important to safety
Maximum Flood Elevation	0.30 m (1 ft) below plant grade in the vicinity of the SSCs important to safety
Precipitation	
- Maximum precipitation rate (1 mi^2)	492.7 mm (19.4 in.) over 1 hour 157.5 mm (6.2 in.) in 5 minutes
- 100-year snowpack roof load	2.873 kPa (60 lbf/ft^2)
- Extreme winter precipitation roof load	5.985 kPa (125 lbf/ft^2)
- Depth of 48-hour probable maximum winter precipitation (PMWP)	914.4 mm (36 in.)
HVAC Outdoor Design Temperature(5)	
- 5 % exceedance values	
· Maximum	35.0 °C (95 °F) dry bulb and 25.0 °C (77 °F) coincident wet bulb
· Minimum	-20.6 °C (-5 °F) dry bulb
- 1 % exceedance values	
· Maximum	37.8 °C (100 °F) dry bulb and 25.0 °C (77 °F) coincident wet bulb
· Minimum	–23.3 °C (–10 °F) dry bulb
- 0 % exceedance values (historical limit excluding peaks < 2 hours)	
· Maximum	46.1 °C (115 °F) dry bulb and 26.7 °C (80 °F) coincident wet bulb
· Minimum	–40.0 °C (–40 °F) dry bulb
Ambient Design Temperature for Cooling Tower	
- Ambient 5 % exceedance values for circulating water system (CWS)	
· Maximum	26.1 °C (79 °F) non-coincident wet bulb
· Minimum	–20.6 °C (-5 °F) dry bulb
- Ambient 0 % exceedance values for essential service water system (ESWS)(5) (historical limit excluding peaks < 2 hours)	
· Maximum	27.2 °C (81 °F) non-coincident wet bulb
· Minimum	–40.0 °C (–40 °F) dry bulb

Attachment

➢ Table 2.0-1 Site Parameters (2/4)

Parameter Description	Parameter Value
Extreme Wind - 50-year 3-second wind gust speed - Importance factor	 64.8 m/s (145 mph) 1.15 (2)
Tornado Parameters - Maximum horizontal wind speed - Translational speed - Rotational speed - Radius of maximum rotational speed - Maximum pressure differential - Rate of pressure drop - Missile spectra	 102.8 m/s (230 mph) 20.6 m/s (46 mph) 82.2 m/s (184 mph) 45.7 m (150 ft) 8.274 kPa (1.2 psi) 3.447 kPa/s (0.5 psi/s) Table 2 (Region I) of NRC RG 1.76 (Reference 1)
Hurricane Parameters - Maximum 3-second wind gust speed - Missile spectra	 116 m/s (260 mph) Table 1 of NRC RG 1.221 (Reference 2)
Accident Release χ/Q Values at exclusion area boundary (EAB) · 0-2 hr	 1.00×10^{-3} s/m^3
Accident Release χ/Q Values at low-population zone (LPZ) · 0-8 hr · 8-24 hr · 24-96 hr · 96-720 hr	 2.20×10^{-4} s/m^3 1.60×10^{-4} s/m^3 1.00×10^{-4} s/m^3 8.00×10^{-5} s/m^3
Annual Average χ/Q Values at Site Boundary · Undepleted/no decay · Undepleted/2.26-day decay · Depleted/8.00-day decay · Relative deposition factor (D/Q)	 2.00×10^{-5} s/m^3 1.99×10^{-5} s/m^3 1.84×10^{-5} s/m^3 2.00×10^{-7} 1/m^2
Inventory of radionuclides that could seep into the groundwater	See Table 11.2-9
Safe Shutdown Earthquake (SSE)	0.3 g peak ground acceleration

Attachment

➢ Table 2.0-1 Site Parameters (3/4)

Parameter Description	Parameter Value
Certified Seismic Design Response Spectra (CSDRS) Referencing SSE	See Figures 2.0-1 and 2.0-2
Hard Rock High Frequency (HRHF) Response Spectra[(4)]	0.46g peak ground acceleration See Figures 2.0-3 and 2.0-4
Tectonic and Non-tectonic Surface Deformation Potential	See Subsection 2.5.3
Allowable Static Bearing Capacity	The allowable static bearing capacity, including a factor of safety appropriate for the design load combinations, shall be greater than or equal to the maximum static bearing demand of 718.2 kPa (15 ksf). The allowable static bearing capacity is the value of ultimate bearing capacity divided by 3.0.
Allowable Dynamic Bearing Capacity	The allowable dynamic bearing capacity, including a factor of safety appropriate for the design load combinations, shall be greater than or equal to the maximum dynamic bearing demand of 2,872.8 kPa (60 ksf). The allowable dynamic bearing capacity is the value of ultimate bearing capacity divided by 2.0.
Minimum Factor of Safety for Slope on Static condition	1.5
Minimum Factor of Safety for Slope on Dynamic condition (SSE)	1.2
Minimum Shear Wave Velocity	304.8 m/s (1,000 ft/s)
Maximum Dip Angle for Soil Uniformity	20 degrees
Liquefaction Potential	See Subsection 2.5.4.8
Maximum Allowable Differential Settlement inside Building	12.7 mm (0.5 in.) per 15.24 m (50 ft) in any direction for the Seismic Category I structures
Maximum Allowable Differential Settlement between Buildings	76.2 mm (3.0 in.) between NI Common Basemat and EDG Building & DFOT Building 12.7 mm (0.5 in.) between other adjacent buildings
Minimum Soil Angle of Internal Friction	Greater than or equal to 35 degrees below the footprint of the seismic Category I structures at their excavation depth
Slope Failure Potential (yes/no)	No
Backfill Material Density	2.2 g/cm^3 (137 pcf)

Attachment

➢ Table 2.0-1 Site Parameters (4/4)

Parameter Description	Parameter Value		
Backfill Material Dynamic Poisson's Ratio	0.33		
Backfill Material Dynamic Properties (Normalized Shear Moduli & Maximum Damping) (6)	Shear Strain(%)	G/G_{max}	Damping (%)
	1.0	0.05	24.0
	0.1	0.22	16.0
	0.01	0.54	6.0
	0.001	0.85	2.0
	0.0001	1.00	1.0
Strain-compatible Minimum Shear-wave Velocity of Backfill	155 m/s (510 fps)		

(1) Plant grade represents the level of ground adjacent to the nuclear island buildings and is established at a plant elevation 98 ft 8 in.

(2) 100-year recurrence interval: Value to be used for design of seismic Category I and II structures only.

(3) Bearing capacity is defined at the foundation level of the seismic Category I structures.

(4) The HRHF response spectra are provided for evaluation of site-specific ground motion response spectra which exceed the CSDRS in the high frequency range at hard rock sites.

(5) Degrees Fahrenheit (°F) are the main units and temperatures in Celsius (°C) are reference values that are converted from main units.

(6) The backfill material dynamic properties are used to calculate the shear-straincompatible shear wave velocity profiles for the backfill. The strain-compatible damping values of the backfill cannot be greater than 15%.

Attachment

➢ List of COL Items (1/5)

COL No.	Description
COL 2.0(1)	The COL applicant is to demonstrate that the APR1400 design meets the requirements imposed by the site-specific parameters and conforms with all design commitments and acceptance criteria if the characteristics of the site fall outside the assumed site parameters in Table 2.0-1.
COL 2.1(1)	The COL applicant is to provide site-specific information on the site location and description of the site, exclusion authority and control, and population distribution as stated in NRC RG 1.206.
COL 2.2(1)	The COL applicant is to provide site-specific information on nearby industrial, transportation, and military facilities as required in NRC RG 1.206
COL 2.2(2)	The COL applicant is to identify the DBE caused by nearby industrial, transportation, and military facilities and determine its design parameters.
COL 2.3(1)	The COL applicant is to provide site-specific information on meteorology including regional climatology, local meteorology, onsite meteorological measurement program, estimated short-term atmospheric dispersion for accident release, and long-term atmospheric dispersion estimates for routine release as addressed in NRC RG 1.206.
COL 2.3(2)	The COL applicant is to perform the radiological consequence analysis and demonstrate that the related dose limits specified in 10 CFR 50.34 and GDC 19 are not exceeded, if the site-specific χ/Q values exceed the bounding values described in Tables 2.3-1 to 2.3-12.
COL 2.4(1)	The COL applicant is to provide the site-specific hydrologic information on probable maximum precipitation (PMP), probable maximum flood (PMF) of streams and rivers, potential dam failures, probable maximum surge and seiche flooding, probable maximum tsunami hazards, ice effects, cooling water canals and reservoirs, channel diversions, flood protection requirements, low water considerations, groundwater, potential accidental release of liquid effluents in ground and surface water, and Technical Specifications and emergency operation requirements in accordance with NRC RG 1.206, NRC RG 1.59, and NRC JLD-ISG-2012-06.

Attachment

List of COL Items (2/5)

COL No.	Description
COL 2.5(1)	The COL applicant is to provide the site-specific information on geology, seismology, and geotechnical engineering as required in NRC RG 1.206. The COL applicant is to conduct the probabilistic seismic hazard analysis (PSHA) and develop the site-specific GMRS using the PSHA results as required in NRC RG 1.208.
COL 2.5(2)	The COL applicant is to confirm that the FIRS of the nuclear island are completely enveloped by the CSDRS-compatible free-field response motions at the bottom elevation of the nuclear island for a site with the low-strain shear wave velocity greater than 304.8 m/s (1,000 ft/s) at the finished grade in the free field. Alternately, the COL applicant is to confirm that the FIRS of the nuclear island are completely enveloped by the CSDRS for a hard rock site with a low-strain shear wave velocity of supporting medium for the nuclear island greater than 2,804 m/s (9,200 ft/s).
COL 2.5(3)	The COL applicant is to confirm that (i) the requirement for the site-specific weight densities of subsurface soils is to be no less than 2,002.3 kg/m3 (125 lb/ft^3) and the site-specific strain-compatible soil hysteresis damping ratio profile is to be equal to or less than that shown in Table 3.7A-1; (ii) the profiles of site-specific soil properties (weight density, strain compatible soil shear and compression wave velocity, and strain-compatible soil hysteresis damping ratio) are generally increasing with depth from the ground surface in a manner similar to the general profile shapes shown in Tables 3.7A-1 through 3.7A-9 and Figures 3.7A-3 through 3.7A-11; (iii) the site-specific soil profiles have no inverse condition, i.e., the soil properties of a deeper soil layer are less than the properties of the soil layer above it; and (iv) the site-specific strain-compatible soil shear wave velocity profile, including backfill, is consistent with one of the APR1400 generic site conditions S1 through S9 considered for the standard design as shown in Tables 3.7A-1 through 3.7A-9 and Figures 3.7A-3 through 3.7A-11.

Attachment

➢ List of COL Items (3/5)

COL No.	Description
COL 2.5(3) (cont'd)	The site specific upper bound and lower bound low-strain shear wave velocity profiles are to be bounded by the APR1400 generic low-strain shear wave velocity profiles S1 through S9 considered for the standard design. The lower bound and upper bound shear wave velocity profiles correspond to the Gmax/1.5 and 1.5Gmax of the site-specific low-strain soil shear modulus profile where Gmax is the low-strain maximum shear modulus. The lower bound low-strain shear wave velocity for the site-specific soil profile is not to be less than 304.8 m/s (1,000 ft/s)
COL 2.5(4)	The COL applicant is to confirm that the site-specific profile for a HRHF site is consistent with generic soil profile S9 and the site-specific GMRS determined at the finished grade are completely enveloped by the APR1400 HRHF response spectra shown in Figures 3.7-12 and 3.7-13. In addition, the COL applicant is to confirm that the FIRS of the seismic Category I structures are completely enveloped by the HRHF-compatible free-field response motions at the bottom elevation of each seismic Category I structure.
COL 2.5(5)	The COL applicant is to perform a site-specific seismic analysis to develop in-structure response spectra at key locations, strain-compatible properties, embedment and extent of backfill using the procedure described in Appendix 3.7A if COL 2.5(2) and COL 2.5(3) above are not met. In addition, the COL applicant is to confirm that the site-specific 5% damped in-structure response spectra so generated are enveloped by the corresponding 5% damped in-structure response spectra provided in Appendix 3.7A.
COL 2.5(6)	The COL applicant is to perform a site-specific seismic response analysis using the procedure described in Appendix 3.7B and the EPRI White Paper "Seismic Screening of Components Sensitive to High Frequency Vibratory Motions" (Reference 6), if COL 2.5(4) is not met.

Attachment

➢ List of COL Items (4/5)

COL No.	Description
COL 2.5(7)	The COL applicant is to perform an evaluation of the subsurface conditions within the standard plant structure footprint based on the geologic investigation in accordance with NRC RG 1.132.
COL 2.5(8)	The COL applicant is to confirm that the dynamic properties of SFG to be used in construction of the APR1400 seismic Category I structures and compressive strength of lean concrete under the nuclear island basemat satisfy the SFG requirements provided in Table 2.0-1 and minimum compressive strength of 140 kg/cm2 (2,000 psi).
COL 2.5(9)	The COL applicant is to evaluate the potential future surface deformation of tectonic and non-tectonic origin.
COL 2.5(10)	If the potential for future surface deformation exists, the COL applicant needs to demonstrate the potential effects of surface deformation are within the design basis of the facility.
COL 2.5(11)	The COL applicant will evaluate the allowable bearing capacity of the subsurface based on the site-specific properties of the underlying materials, including appropriate laboratory test data to evaluate strength, and considering local site effects, such as fracture spacing, variability in properties, and evidence of shear zones. If the site-specific allowable bearing capacity is less than the maximum bearing demands specified in Table 2.0-1, a site-specific evaluation shall be performed by a COL applicant using the APR1400 basemat model and methodology described in subsection 3.8.5.
COL 2.5(12)	The COL applicant will verify whether the predicted settlement exceeds settlements within each building and between buildings exceed the maximum differential settlement within building specified in Table 2.0-1 or not. If the predicted settlement exceeds the maximum value in Table 2.0-1, a detailed site specific evaluation shall be performed by a COL applicant using the APR1400 basemat model and methodology described in subsection 3.8.5 to demonstrate acceptable.

Attachment

➢ List of COL Items (5/5)

COL No.	Description
COL 2.5(13)	The COL applicant is to evaluate the potential for liquefaction occurring at the site adjacent to and under seismic Category I structures in accordance with NRC RG 1.198. In addition, the COL applicant is to evaluate the liquefaction potential for those seismic Category II structure foundations that if failed, could degrade the function of a seismic Category I SSC to an unacceptable safety level.
COL 2.5(14)	The COL applicant is to provide site-specific information about the static and dynamic stability of all natural and man-made soil and rock slopes including embankments and dams.
COL 2.5(15)	The COL applicant is to confirm that the soil angle of internal friction below the footprint of the seismic Category I structures at their excavation depth is a minimum of 35 degrees.

Presentation to the ACRS Subcommittee

Korea Hydro Nuclear Power Co., Ltd (KHNP)

APR1400 Design Certification Application Review

Safety Evaluation Report with open Items

Chapter 2: SITE CHARACTERISTICS

September 21, 2016

Presentation Contents

- Staff Review Team
- Overview of Design Certification Application
- Technical Areas of Interest
  - 2.1 Geography and Demography
  - 2.2 Nearby Industrial, Transportation, and Military Facilities
  - 2.3 Meteorology (will be presented on October 4, 2016)
  - 2.4 Hydrologic Engineering
  - 2.5 Geology, Seismology, and Geotechnical Engineering

ACRONYMS

- ADAMS- Agencywide Documents Access & Management System
- CFR- Code of Federal Regulation
- COL- Combined License
- DC - Design Certification
- DBA- Design Basis Accident
- DBE- Design Basis Event
- DCD- Design Control Document
- D/Q- Relative Deposition Factor ($1/m^2$)
- EAB – Exclusion Area Boundary
- EPRI- Electric Power Research Institute
- ESP- Early Site Permits
- FSAR- Final Safety Analysis Report
- GDC-General Design Criterion

ACRONYMS

- ISG- Interim Staff Guidance
- ITAAC- Inspections, tests, analyses, and acceptance criteria
- KHNP- Korea Hydro & Nuclear Power
- LPZ- Low Population Zone
- OI- Open Item
- PMWP- Probable Maximum Winter Precipitation
- RAI- Request for Additional Information
- RG- Regulatory Guide
- SAR- Safety Analysis Report
- SE- Safety Evaluation
- SRP- Standard Review Plan
- SSC- Structures Systems and Components
- URD- Utility Requirements Documents
- χ/Q – Atmospheric Dispersion Factor (sec/m^3)

- **Technical Staff Presenters**
  - **Seshagiri Tammara- DCD Section 2.1 and 2.2** Radiation Protection and Accident Consequences Branch
  - **Michael Lee- DCD Section 2.4**
  - Hydrology & Meteorology Branch
  - **Ricardo Rodriguez- DCD Section 2.5**

    Geoscience and Geotechnical Engineering Branch
- **Project Managers**
  - **Jeff Ciocco – Lead PM**
  - **Tarun Roy– Chapter PM**

Staff Review Team

- **Seshagiri Tammara**
  Radiation Protection and Accident Consequences Branch
- **Michael Lee**
  Hydrology and Meteorology Branch
- **Ricardo Rodriguez**
  Geoscience and Geotechnical Branch
- **Stephanie Devlin-Gill**
  Geoscience and Geotechnical Branch
- **David Heeszel**
  Geoscience and Geotechnical Branch
- **Luissette Candelario**
  Geoscience and Geotechnical Branch

Overview of Design Certification Application

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

SRP Section/Application Section		Number of RAI Questions	Number of SE Open Items
2.0	Site Characteristics	0	0
2.1	Geography and Demography	0	0
2.2	Nearby Industrial, Transportation, and Military Facilities	0	0
2.3	Meteorology	10	0
2.4	Hydrologic Engineering	0	0
2.5	Geology, Seismology, and Geotechnical Engineering	23	0
Totals		33	0

Section 2.1 - Geography and Demography

- The review involves the following sections of the KHNP APR 1400 DCD:
  - 2.1.1 - Site Location and Description
  - 2.1.2 - Exclusion Area Authority and Control
  - 2.1.3 - Population Distribution

The COL applicant is to provide this site specific information as part of COL information Item 2.1(1) in the COL application.

Section 2.2 - Nearby Industrial, Transportation, and Military Facilities

- The review involves the following sections of the KHNP APR 1400 DCD:
  - 2.2.1 - Locations and Routes
  - 2.2.2 - Descriptions
  - 2.2.3 - Evaluation of Potential Accidents

The COL applicant is to provide this site specific information as part of COL information Item 2.2(1) and COL information Item 2.2(2) in the COL application.

United States Nuclear Regulatory Commission
Protecting People and the Environment

Section 2.4 – Hydrologic Engineering

- The review involves the following sections of the APR 1400 DCD:
  - 2.4.1 Hydrologic Description
  - 2.4.2 Floods
  - 2.4.3 Probable Maximum Flood on Streams and Rivers
  - 2.4.4 Potential Dam Failures
  - 2.4.5 Probable Maximum Surge and Seiche Flooding
  - 2.4.6 Probable Maximum Tsunami Hazards
  - 2.4.7 Ice Effects
  - 2.4.8 Cooling Water Canals and Reservoirs
  - 2.4.9 Channel diversion
  - 2.4.10 Flooding Protection Requirements
  - 2.4.11 Low Water Considerations
  - 2.4.12 Groundwater
  - 2.4.13 Accidental Release of Radioactive Liquid Effluents in Ground and Surface Water
  - 2.4.14 Technical Specifications and Emergency Operations Requirements
  - 2.4.15 Combined License Information

Technical Topics of Interest
Section 2.4 -Hydrologic Engineering

- Information in all the sections is site specific and will be provided by the COL applicant.
- Hydrologic Parameters
  - The DCD applicant identified the following hydrologic parameters:
    - Maximum groundwater level (2 ft below finished grade)
    - Maximum flood (tsunami) level (1 ft below finished grade)
    - Maximum hourly rainfall rate (19.4 in/hr.)
    - Maximum short-term rainfall rate (6.2 in/5 min.)
- A COL applicant needs to demonstrate that its site characteristics fall within the APR 1400 DCD site parameters
- Staff previously evaluated these parameters to determine the acceptability of the parameter values

Technical Topics of Interest
Section 2.4 -Hydrologic Engineering

- COL Information Item 2.4(1)
  - COL applicant is to provide sufficient information to verify that hydrologic-related events will not affect the safety-basis for the APR 1400
  - COL applicant is to characterize the potential flooding hazards that may affect proposed reactor site
  - COL applicant decides on what combination of design features (i.e., elevation of door openings, the addition of fill, passive drainage systems, berms, administrative procedures) will be used to mitigate effects of any potential flooding hazard proposed reactor site.

Conclusion
Section 2.4 – Hydrologic Engineering

- The DCD Applicant has provided plant specific hydrological site parameters and the Staff finds those acceptable.
- The DCD Applicant has properly identified the site specific information to be provided as part of any future COL application.
- The Applicant has satisfactorily answered all earlier RAI's and there are no open items.

Technical Topics of Interest

Section 2.5 – Geology, Seismology, and Geotechnical Engineering

- The review involved the following sections of the APR 1400 DCD:
    - 2.5.1 Basic Geologic and Seismic Information
    - 2.5.2 Vibratory Ground Motion
    - 2.5.3 Surface Deformation
    - 2.5.4 Stability of Subsurface Materials and Foundations
    - 2.5.5 Stability of Slopes

Technical Topics of Interest

Section 2.5 – Geology, Seismology, and Geotechnical Engineering

- Establishes site parameters for a candidate site, including:
  - Maximum allowable bearing demand
  - Minimum shear wave velocity
  - Allowable settlements (total, differential)
  - No liquefaction potential for seismic category I structures
  - No surface tectonic deformation potential within the exclusion area boundary

- These parameters are expected to be representative of a potential COL or ESP site.

Technical Topics of Interest
Section 2.5 – Geology, Seismology, and Geotechnical Engineering

- Sections 2.5.1, 2.5.2 and 2.5.3 findings:
  - DCD requires that site specific geology considerations are addressed by COL applicants in accordance with applicable regulations
  - DCD requires COL applicants to conduct site specific PSHA and site response analysis in accordance with relevant regulations
  - Site specific subsurface profiles are compared to generic profiles in DCD to ensure that parameters in DCD Chapter 3.7 analyses are relevant to COL sites
  - DCD requires that COL applicants show no potential for tectonic or non-tectonic surface deformation at the site

Technical Topics of Interest
Section 2.5 – Geology, Seismology, and Geotechnical Engineering

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- Sections 2.5.4 and 2.5.5 findings:
  - The majority of staff RAIs focused on:
    - Soil or rock uniformity requirements
    - Applicability of static and dynamic parameters of the subsurface, including the SFG (structural fill granular) and lean concrete backfill
    - Clarifications regarding the calculation of bearing capacity and settlement for safety related structures
    - Liquefaction potential adjacent to and under Seismic Category I structures
    - Static and dynamic stability of all natural and man made slopes
- The applicant adequately addressed all RAIs. In many instances, tasked any future COL applicant to address pertinent issues during the COL application process through Information Items

Technical Topics of Interest

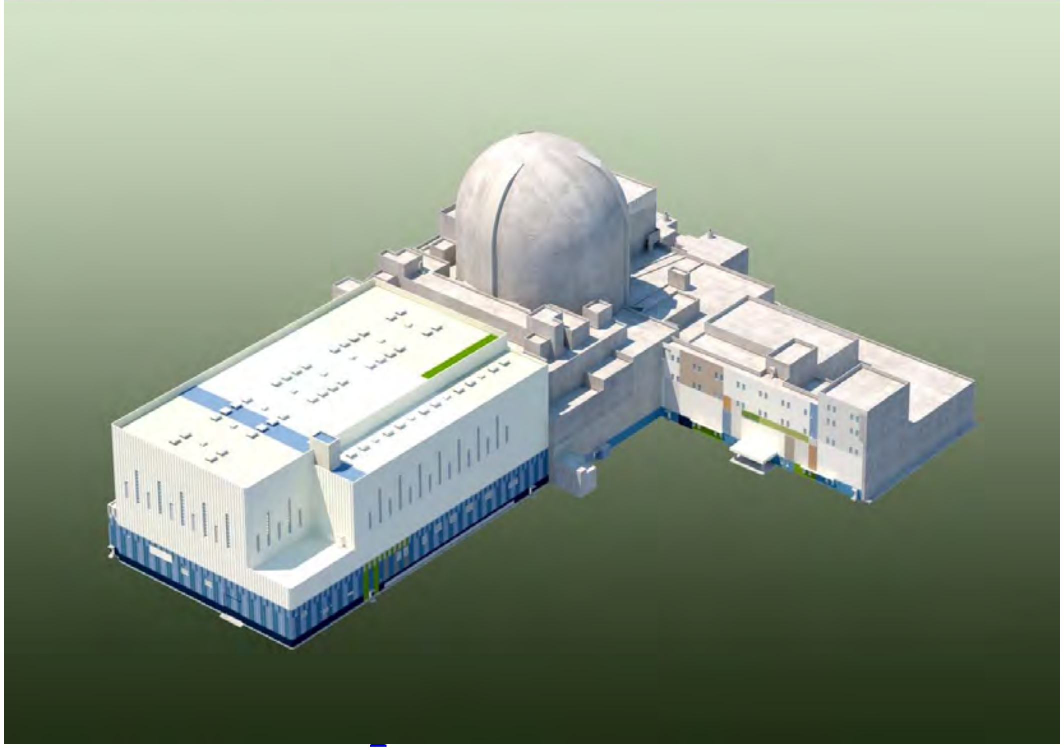
Section 2.5 – Geology, Seismology, and Geotechnical Engineering

<u>Staff Conclusions</u>

- No Open Items identified in Chapter 2.5
- 15 Information Items identified in Chapter 2.5 adequately represent the actions need to be taken by potential COL applicants
- The applicant properly identified appropriate geologic, seismologic and geotechnical site parameters and properly identified the site specific information to be provided as part of any future COL application.
- 10 Confirmatory Items are currently being tracked by the staff

Questions?

APR1400 DCA
Chapter 5: Reactor Coolant System and Connecting Systems



KEPCO/KHNP

September 21~22, 2016

Contents

Overview of Chapter 5

● List of Submitted Documents

Document No.	Title	Revision	Type	ADAMS Accession No.
APR1400-K-X-FS-14002 -P & NP	APR1400 Design Control Document Tier 2: Chapter 5 Reactor Coolant System and Connecting Systems	0	DCD	ML15006A044
APR1400-K-X-IT-14001 -P & NP	APR1400 Design Control Document Tier 1	0	DCD	ML15006A039
APR1400-Z-A-NR-14015 -P & NP	Neutron Fluence Calculation Methodology for Reactor Vessel	0	TER*	ML15009A124
APR1400-Z-M-NR-14008 -P & NP	Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown	0	TER (IBR**)	ML15009A125
APR1400-A-M-NR-14001 -P & NP	KHNP APR1400 Flywheel Integrity Report	0	TER	ML15009A126
APR1400-E-N-NR-14005 -P & NP	Shutdown Evaluation Report	0	TER	ML15128A285

* TER: Technical Report, **IBR: Incorporated by Reference

● RAI Summary

No. of Questions	No. of Responses*	Not Responded	No. of Open Items
77	77	0	13

* The responses to 6 RAIs for RCP flywheel were submitted after the issue of SE (Sep. 12, 2016).

Overview of Chapter 5

● Section Overview

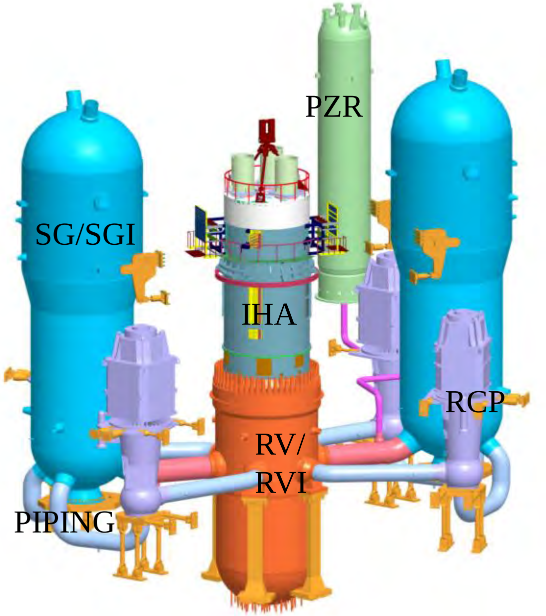
Section	Major Contents	Presenter
5.1 Summary Description	Design Features of the Reactor Coolant System (RCS)	Minseok KIM
5.2 Integrity of Reactor Coolant Boundary (RCPB)	Describes the measures that provide and maintain the integrity of the RCPB throughout the facilities design life	
5.2.1	Conformance with Codes and Code Cases	Minseok KIM
5.2.2	Overpressure Protection	Minseok KIM
5.2.3	Reactor Coolant Pressure Boundary Materials	Jongsoo KIM
5.2.4	In-service Inspection and Testing	Jongsoo KIM
5.2.5	RCPB Leakage Detection	Minseok KIM
5.3 Reactor Vessel	Describes Reactor Vessel Material Issues	
5.3.1	Reactor Vessel Materials	Jongsoo KIM
5.3.2	P-T limits, Pressurized Thermal Shock, Upper-Shelf Energy	Seognam CHOI
5.3.3	Reactor Vessel Integrity	Jongsoo KIM
5.4 RCS Component and Subsystem Design	Describes the Reactor System Components	
5.4.1	Reactor Coolant Pumps (RCPs)	Minseok KIM
5.4.2	Steam Generators (SGs)	Seognam CHOI
5.4.7	Shutdown Cooling System (SCS)	Minseok KIM
5.4.10	Pressurizer	Seognam CHOI
5.4.12	RCS High Point Vents	Minseok KIM
5.4.15	Component Supports	Seognam CHOI

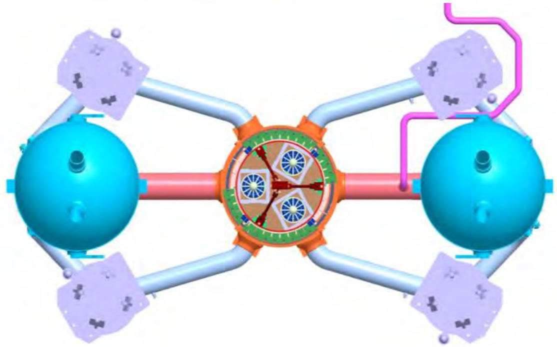
Reactor Coolant System and Connecting Systems

● 5.1 Summary Description (1/3)

➢ **Design Features of APR1400**

- **Design Life:** 60 Years
- **Power:** 4,000 MWth / 1,400 MWe
- **Two Loops:** Symmetric
  - ✓ 1 RV, 2 SGs, 4 RCPs, 1 Pressurizer
  - ✓ 2 Hot Legs, 4 Cold Legs
- **Primary Operating Condition:**
  - ✓ Pressure: 2,250 psia
  - ✓ NOP Hot / Cold Temp.: 615 / 555 ^{o}F
- **Secondary Operating Condition:**
  - ✓ Pressure: 1,000 psia
  - ✓ MF/MS Temp.: 450 / 545 ^{o}F
- **SG Tube:** 13,102 /SG, Alloy 690TT
- **Pressurizer:** 2,400 ft^3





Reactor Coolant System and Connecting Systems

● 5.1 Summary Description (2/3)

➢ RCS Functions

- Energy transfer from the reactor core to the steam generator
- Secondary barrier to the release of fission products
- Sufficient cooling during all normal plant operations and expected transients
- Reactor coolant circulation with the required chemistry & boron concentration
- System pressure control to a moderate extent through sprays
- Rapid depressurization of the RCS by manual operation of POSRVs
- Venting of steam and non-condensable gases

Reactor Coolant System and Connecting Systems

● 5.1 Summary Description (3/3)

➢ **Principal Parameters of the RCS (Table 5.1.1-1)**

Parameter	Value
Design thermal power, MWt (including net heat addition from pumps)	4,000
Design pressure, kg/cm^2A (psia)	175.8 (2,500)
Design temperature (except pressurizer), °C (°F)	343.3 (650)
Pressurizer design temperature, °C (°F)	371.1 (700)
Coolant flow rate, kg/hr (lbm/hr)	75.6×10^6 (166.6×10^6)
Cold leg temperature, operating, °C (°F)	290.6 (555)
Average temperature, operating, °C (°F)	307.2 (585)
Hot leg temperature, operating, °C (°F)	323.9 (615)
Normal operating pressure, kg/cm^2A (psia)	158.2 (2,250)
System water volume, m^3 (ft^3) (with pressurizer)	455.3 (16,079)
Pressurizer water volume, m^3 (ft^3) (full power)	33.2 (1,171)
Pressurizer steam volume, m^3 (ft^3) (full power)	35.7 (1,260)

Reactor Coolant System and Connecting Systems

● 5.2.1 Conformance with Codes and Code Cases

➢ Conformance with 10 CFR 50.55a

- ◆ Construction: ASME Section III 2007 Edition with 2008 Addenda
  - ✓ RCPB components - Class 1 components
  - ✓ RCPB components that meet the exclusion requirements - Class 2 components
  - ✓ The remaining safety-related components - Class 3 components
- ◆ ISI: ASME Section XI 2007 Edition with 2008 Addenda (Sections 5.2.4 and 6.6)
- ◆ IST: ASME OM Code 2004 Edition with 2005 and 2006 Addenda – Section 3.9.6

➢ Compliance with Applicable Code Cases

- ◆ RG 1.84, Design, Fabrication and Materials Code Case Acceptability, ASME Section III, Rev. 36, U.S. Nuclear Regulatory Commission, August 2014.
- ◆ RG 1.147, In-service Inspection Code Case Acceptability, ASME Section XI, Division 1, Rev. 17, U.S. Nuclear Regulatory Commission, August 2014.
- ◆ RG 1.192, Operation and Maintenance Code Case Acceptability, ASME OM Code, Rev. 1, U.S. Nuclear Regulatory Commission, August 2014.

Reactor Coolant System and Connecting Systems

● 5.2.2 Overpressure Protection (1/4)

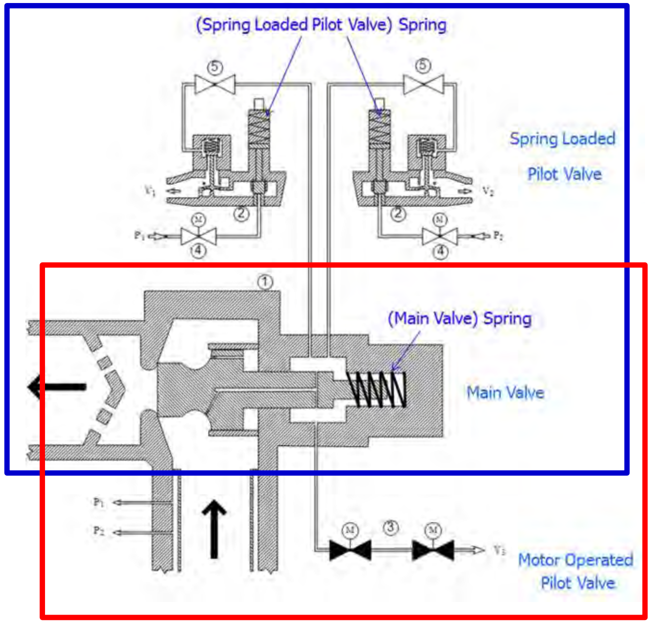
- **Overpressure Protection of the RCS and Steam Generators**
  - Pilot-operated safety relief valves (POSRVs)
  - Main steam safety valves (MSSVs)
  - Reactor protection system (RPS)
- **Maintain the RCS pressure below 110 percent of design pressure during the worst-case loss-of-load event with a delayed reactor trip**
- **Low Temperature Overpressure Protection (LTOP)**
  - Shutdown cooling system (SCS) suction line relief valve

Reactor Coolant System and Connecting Systems

● 5.2.2 Overpressure Protection (2/4)

➢ Pressurizer POSRV

- Main Valve (1)
- Spring Loaded Pilot Valve (SLPV) (2)
  - ✓ Automatic actuation for RCS overpressure protection
- Motor Operated Pilot Valve (2)
  - ✓ Manual actuation for RCS rapid depressurization
- Pilot-Operated Pressure Relief Valves are Permitted by ASME Sec.III NB-7174 as Pressure Relief Device
  - ✓ Meeting the requirements of NB-7520
- ASME Rated Capacity (540,000 lbs/hr) for Saturated Steam according to NB-7734
  - ✓ Ex) ASME certification number: 64066
- Section 15.2 provides the Functional Design Evaluation of the Overpressurization Protection System.



M: Motor Operated Valve
Vi: Pilot Discharge
Pi: Impulse Line

Reactor Coolant System and Connecting Systems

● 5.2.2 Overpressure Protection (3/4)

- **Open Item (RAI No. 8244, Question 05.02.02-1, b/c), Date issued (10/2/15); KHNP responded (12/11/15)**
  - **Description of Issue** : Provide the description of the LTOP analysis and, more specifically provide the mass and energy addition transient results including the maximum ΔT.
  - **Resolution :** Provided the description about the mass and energy addition event, transient results, and the assumption for the maximum ΔT.
- **Open Item (RAI No. 8609, Question 05.02.02-7), Date issued (5/18/16); KHNP responded (6/8/16)**
  - **Description of Issue** : Provide the description about LTOP analysis methodology, input parameters, and assumption.
  - **Resolution :** Provided the description about detailed methodology, input parameters, and assumption. And provided why these inputs and assumption are suitably conservative.

Reactor Coolant System and Connecting Systems

● 5.2.2 Overpressure Protection (4/4)

➢ **Open Item (RAI No. 8244, Question 05.02.02-5), Date issued (10/2/15); KHNP responded (6/30/16)**

- **Description of Issue** : Address the difference between the pre-service testing in DCD Section 5.2.2.10 and Chapter 14 test for pressurizer POSRVs.
- **Resolution :** Revised test plans in Section 14.2 to describe the complete pre-service testing for pressurizer POSRVs.

Reactor Coolant System and Connecting Systems

● 5.2.3 Reactor Coolant Pressure Boundary Materials (1/5)

➢ Material Specification and Selection

- ◆ Conform to ASME Section III NB for Class 1 components and RG 1.84

➢ Low-alloy Steel and Carbon-steel

- ◆ RCS components fabricated from low-alloy and carbon steel forgings with cladding
- ◆ Conform to RG 1.43 for clad integrity
- ◆ Conform to 10 CFR 50 Appendix G for fracture toughness properties
- ◆ RT_{NDT} control ≤ 10°F for RCPB, ≤ -10°F for RPV belt line materials
- ◆ Sulfur content limited to 0.010% max.
- ◆ Tight control on chemical elements for RPV belt line to reduce the neutron embrittlement
  - ✓ Cu: 0.05% / 0.03% (welds / forgings), Ni: 0.10% / 1.00% (welds / forgings), P: 0.012%, V: 0.030%, S: 0.010%
- ◆ Major structural materials: SA-508 Grade 3 Class 1, SA-508 Grade 1A

Reactor Coolant System and Connecting Systems

● 5.2.3 Reactor Coolant Pressure Boundary Materials (2/5)

➢ Austenitic Stainless Steel

- Control of use of sensitized stainless steel (comply with RG 1.44)
- Solution annealed condition
- Reduced carbon content of base metals and welding materials (less than 0.065% or L-grade)
- No cold worked austenitic stainless steel for reactor coolant contact parts
- Limited cobalt content (0.05%)

➢ Ni-Cr-Fe Alloys (Alloy 690)

- Control on chemistry and mechanical properties with state-of-the-art industrial requirements
- Thermally treated (TT) after solution annealing
- Control on fabrication practices to limit residual cold work
- Limited cobalt content (0.014 % ave. / 0.020% max. for SG tube, 0.020% for other materials)
- SG tube, all ICI and CEDM nozzles and small nozzles
- Welding material of Alloy 52/52M/152 (Neither Alloy 600 nor 82/182 filler material is used.)

Reactor Coolant System and Connecting Systems

● 5.2.3 Reactor Coolant Pressure Boundary Materials (3/5)

➢ Castings

- Delta ferrite content is limited to prevent thermal aging (CF8M ≤ 14% max.)
- Valve body material

➢ Compatibility with Reactor Coolant

- RCS water chemistry is controlled with up-to-date EPRI primary water chemistry guidelines (Rev. 7)
- Comply with RG 1.36 for non-metallic insulation

➢ Fabrication and Processing of Ferritic Materials

- Comply with ASME Section III NB for fabrication and Non Destruction Examination
- Welding procedure & personnel qualified with ASME Sec. III, IX and followings (if applicable)
  - ✓ Comply with RG 1.50 and Appendix D for pre-heat requirement
  - ✓ Hydrogen removing by post heating
  - ✓ Cladding with RG 1.43
  - ✓ Welder qualification of RG 1.71

Reactor Coolant System and Connecting Systems

● 5.2.3 Reactor Coolant Pressure Boundary Materials (4/5)

➢ Fabrication and Processing of Austenitic Stainless Steel

- Avoidance of **s**tress **c**orrosion **c**racking
  - ✓ Solution heat treatment & rapid cooling and control on exposure to specific sensitizing temperature range (800°F-1500 °F)
  - ✓ Cold worked austenitic stainless steel is not used for RCPB
  - ✓ ASTM A262 Practice A or E corrosion test
- Avoidance of sensitization by welding
  - ✓ Welding procedure qualification with RG 1.44
  - ✓ Delta ferrite control on welding material
  - ✓ Weld heat input control
  - ✓ Interpass temperature control
  - ✓ Reduced carbon content
  - ✓ ASTM A262 Practice A or E corrosion test for ensuring non-sensitization
  - ✓ Nb-stabilized austenitic stainless steel (surge line pipe), L-grade (safe end)
- Control on cleaning and contamination during all phase of fabrication
  - ✓ Maintain cleanliness in accordance with NQA-1
  - ✓ Control on chemical elements of component cleaning water

Reactor Coolant System and Connecting Systems

● 5.2.3 Reactor Coolant Pressure Boundary Materials (5/5)

➢ **Open Item (DCD Section 5.2.3.4.1 d. Avoidance of Sensitization)**

- **Description of Issue** : Can not determine whether the use of weld heat input, maximum interpass temperature, and maximum carbon content specified by the applicant will ensure non-sensitization of the materials.
- **Resolution :** The welding procedure specification (WPS) is qualified with ASME Sec. III and IX using the maximum heat input, interpass temperature and reduced carbon content specified in the DCD. In the qualification process, ASTM A262 Practice A or E test is also performed for ensuring non-sensitization of weld metal and weld heat-affected-zone.

  Since the product welding is to be performed under weld heat input, interpass temperature and carbon content less than the maximum values allowed by the qualified WPS, the weld metal and weld heat-affected-zone will not be sensitized severely.

Reactor Coolant System and Connecting Systems

● 5.2.4 Inservice Inspection and Testing

- **Comply with ASME Section XI with 10 CFR 50.55a for PSI and ISI**
  - 10-year inspection interval plan
  - COL applicant is to prepare the PSI and ISI program
  - ASME code cases in accordance with RG 1.147
- **Inspectability of Components**
  - Components designed to allow personnel and equipment to access for required examination in accordance with Sec. XI IWA-1500
- **Qualification of Examination Personnel and Procedure**
  - Personnel qualification with ASME Sec. XI Appendix VII
  - Performance demonstration with ASME Sec. XI Appendix VIII
- **RPV Closure Head Examination with Code Case N-729-1 and 10 CFR 50.55a**
- **Other Inspection Program: Boric Acid Corrosion (BAC) Program**

Reactor Coolant System and Connecting Systems

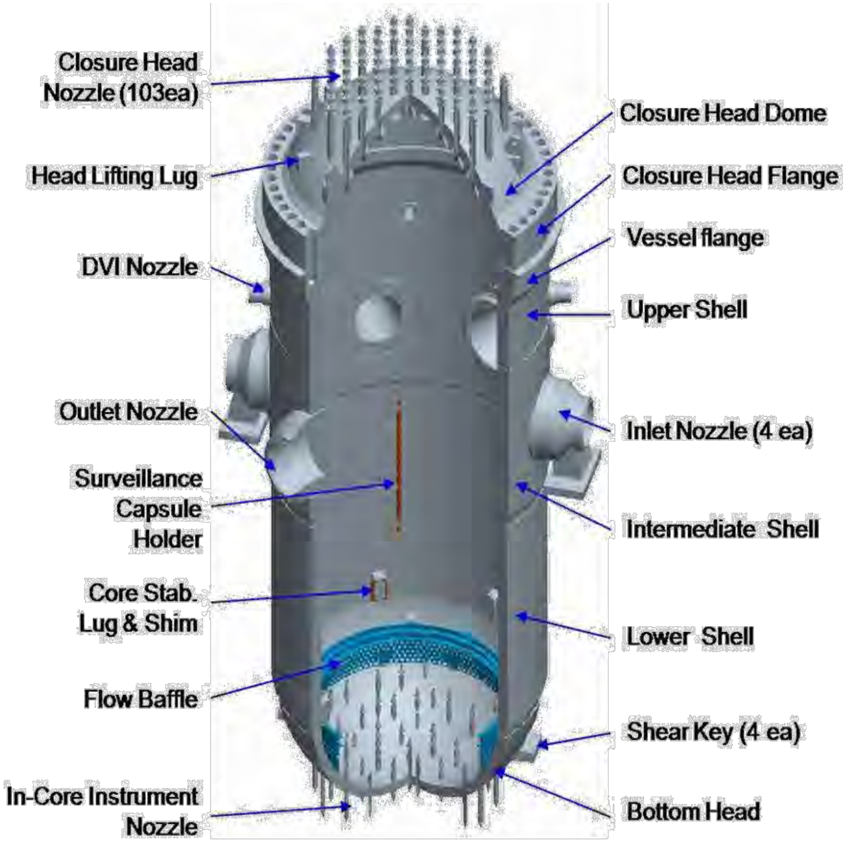
● 5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

- **System Designed in accordance with RG 1.29 and conforms to RG 1.45**
- **Unidentified Leakage**
  - Sensitivity: 0.5 gpm can be detected less than 1 hour
  - Methods: Inventory, sump level and flow, containment air particulate radioactivity monitoring, an acoustic leak monitoring system (ALMS)
- Identified Leakage
  - Definition: (1) Captured/flow-metered/conducted to collection (2) leakage into the containment atmosphere from a known source
  - Leakage instrumentation in the main control room
  - POSRVs, RCP seals, valves, reactor vessel head flange leakage, leakage through steam generator tubes or tubesheet (Chapter 11), leakage to auxiliary systems (Chapter 11)

Reactor Coolant System and Connecting Systems

● 5.3.1 Reactor Vessel Materials

- **Comply with ASME Sec. II, III, V, IX and 10 CFR 50 & Appendices, NRC Reg. Guides**
  - SA-508 Gr. 3 Cl. 1 with stainless steel clad for RPV shells, main nozzles, heads
    - ✓ Initial RT_{NDT} ≤ -10˚F for core belt line, ≤ 10˚F for other RCPB part
    - ✓ Meet the fracture toughness requirements of ASME Sec. III App. G and 10 CFR 50 App. G
  - Alloy 690 for CEDM nozzles and ICI nozzles
  - Surveillance program for RPV in accordance with 10 CFR 50 App. H



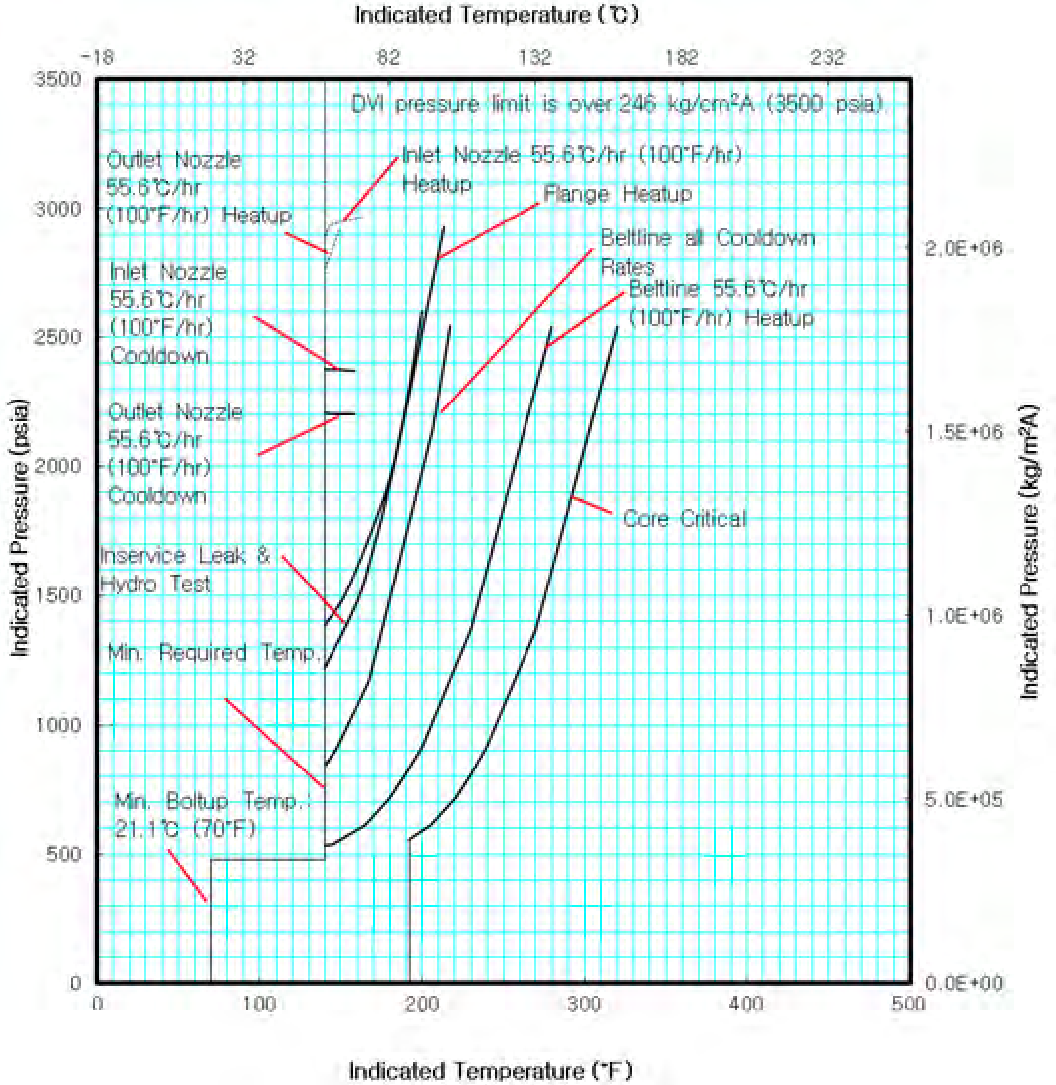
Reactor Coolant System and Connecting Systems

● 5.3.2.1 Pressure-Temperature Limits (1/2)

- **Prevent RCPB materials from non-ductile fracture during heat-up and cool-down transients**
- **Established in accordance with Appendix G of 10 CFR 50 and Appendix G of ASME Code Section XI**
- **Based on end of life (60 years or 55.8 EFPYs based on 93% capacity factor) fast neutron fluence.**
- **Critical Limiting Cases:**
  - RV Beltline: High neutron irradiation induced embrittlement region
  - RV Flange: Bolt preload-induced bending & internal pressure stressed region
  - Inlet, Outlet & Direct Vessel Injection Nozzles: High stressed region

Reactor Coolant System and Connecting Systems

- **5.3.2.1 Pressure-Temperature Limits (2/2)**



Reactor Coolant System and Connecting Systems

● 5.3.2.3 Pressurized Thermal Shock

- RT_{PTS} values at the end of life must be approved by NRC.
- Satisfy the 10 CFR 50.61 Screening Criteria.

Material	Initial RT_{NDT} (℃(℉))	Fluence Factor[1)]	Chemistry Factor[2)] (℃(℉))	ΔRT_{PTS} (℃(℉))	Margin[3)] (℃(℉))	RT_{PTS} (℃(℉))	Screening Criterion (℃(℉))
Lower Shell Course (beltline material)	-23.3 (-10)	1.51	11.1 (20)	16.8 (30.2)	16.8 (30.2)	10.2 (50.4)	132.2 (270)
Weld Material (G-2 & G-3)	-12.2 (10)	1.51	20.8 (37.5)	31.4 (56.6)	31.1 (56)	50.3 (122.6)	148.9 (300)

Notes:

1) Fluence Factor = $f^{(0.28-0.10\log f)}$, fast neutron fluence: $f_{surf} = 9.5 \times 10^{19}$ n/cm^2.

2) Chemistry Factor for Lower shell forging material of 0.03 wt% Cu & 1.00 wt% Ni,
for Weld materials of 0.05 wt% Cu & 0.10 wt% Ni, respectively.

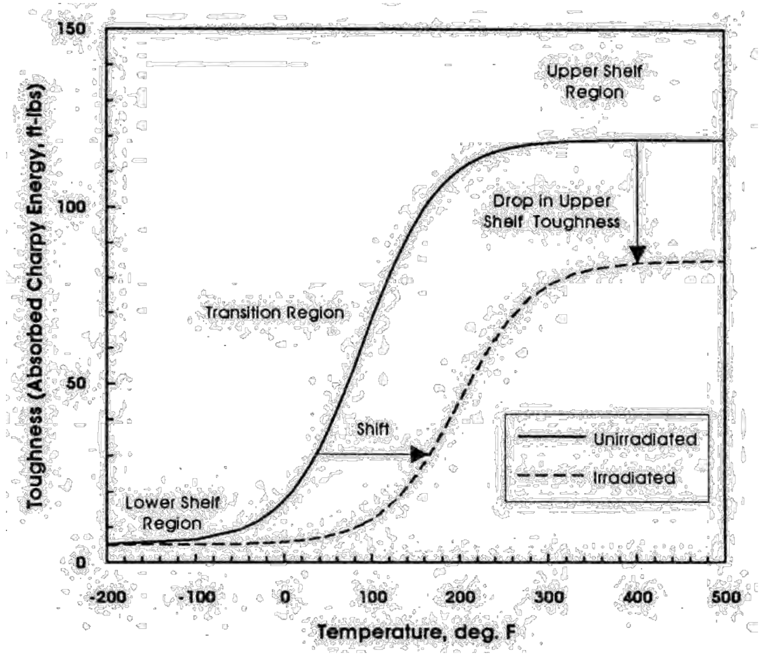
3) Margin = $2(\sigma_u^2+\sigma_\Delta^2)^{0.5}$,
where, $\sigma_u = 0$ and σ_Δ = lesser of 17°F or 0.5 x ΔRT_{PTS} for the beltline material.
$\sigma_u = 0$ and σ_Δ = lesser of 28°F or 0.5 x ΔRT_{PTS} for the weld material.

Reactor Coolant System and Connecting Systems

● 5.3.2.4 Upper-Shelf Energy (USE)

- **Established in accordance with Appendix G of 10 CFR 50 and Branch Tech. Position (BTP) 5-3 of NRC Standard Review Plan**
  - **USE_**Initial, min.: Higher than 102 Joules (75 ft-lbs)
  - **USE_**End of Life, min.: Higher than 68 Joules (50 ft-lbs)

- **With the initial limiting value of 102 Joules (75 ft-lbs), the end-of-life (EOL) USE for APR1400 is estimated to be 69.4 Joules (51 ft-lbs), which is greater than the limiting EOL value.**



Reactor Coolant System and Connecting Systems

● 5.3.3 Reactor Vessel Integrity

- **Proven Fabrication Techniques and Well-characterized Steels are used for RPV Integrity.**
- **Structural Integrity of RPV is provided by meeting the ASME Section III rules for Class 1 Components**
- **Materials, Fabrication Processes, Inspection, Shipment and Installation of RPV are in accordance with ASME Codes and Standards, 10 CFR 50 requirements and applicable Reg. Guides.**
- **Surveillance of RPV Integrity**
  - Inservice inspection program for RPV with ASME Sec. XI and 10 CFR 50.55a
  - Material surveillance program for RPV belt line region with 10 CFR 50 App. H

Reactor Coolant System and Connecting Systems

● 5.4.1 Reactor Coolant Pump (1/7)

Parameter	Value
Number of units	4
Type	Vertical, single-stage centrifugal
Rated total dynamic head, m (ft)	109.7 (360)
Rated flow, L/min (gpm)	460,256 (121,600)
Design pressure, kg/cm^{2}A (psia)	175.8 (2,500)
Design temperature, °C (°F)	343.3 (650)
Normal operating pressure, kg/cm^{2}A (psia)	158.2 (2,250)
Normal operating temperature, °C (°F)	290.6 (555)
NPSH available (at rated flow), m (ft)	152.4 (500)
Suction temperature, °C (°F)	290.6 (555)
Water volume, each, m^3 (ft^3)	3.26 (115)
Weight (including motor), dry, kg (lb)	144,515 (318,600)
Rotating inertia, pump, and motor: Assembly, minimum, kg-m^2 (lbs-ft^2)	6,717 (159,400)
Shaft seals	Mechanical face seals
Pump speed, rpm	1190
Motor synchronous speed, rpm	1200
Motor type	AC induction
Horsepower, hot, kW (hp)	7,457 (10,000)
cold, kW (hp)	10,067 (13,500)
Rated brake horsepower, kW (hp)	7,457 (10,000)
Voltage, V	13200
Phase	3
Frequency, Hz	60
Insulation class	F
Starting current, at 100 % voltage, amps	3426

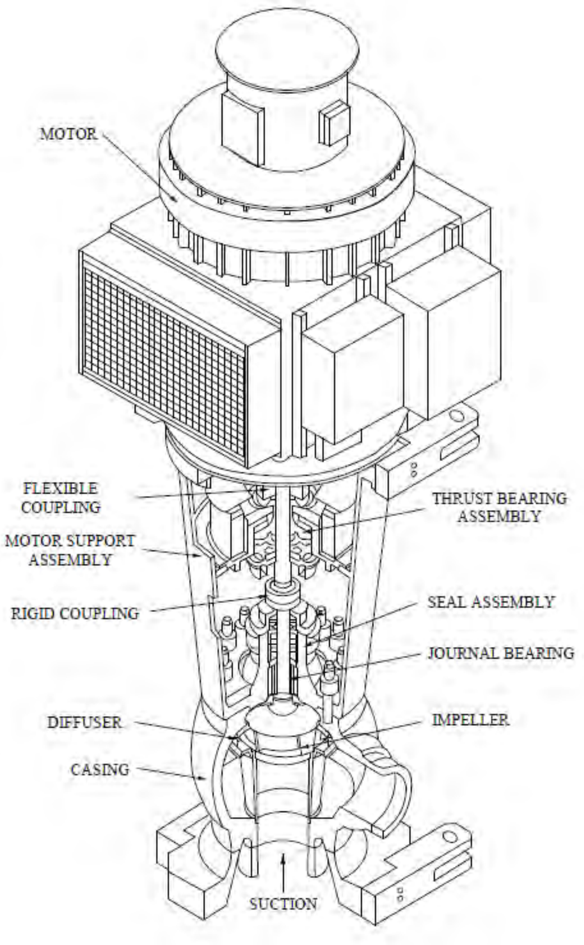


Figure 5.4.1-1 Reactor Coolant Pump

Reactor Coolant System and Connecting Systems

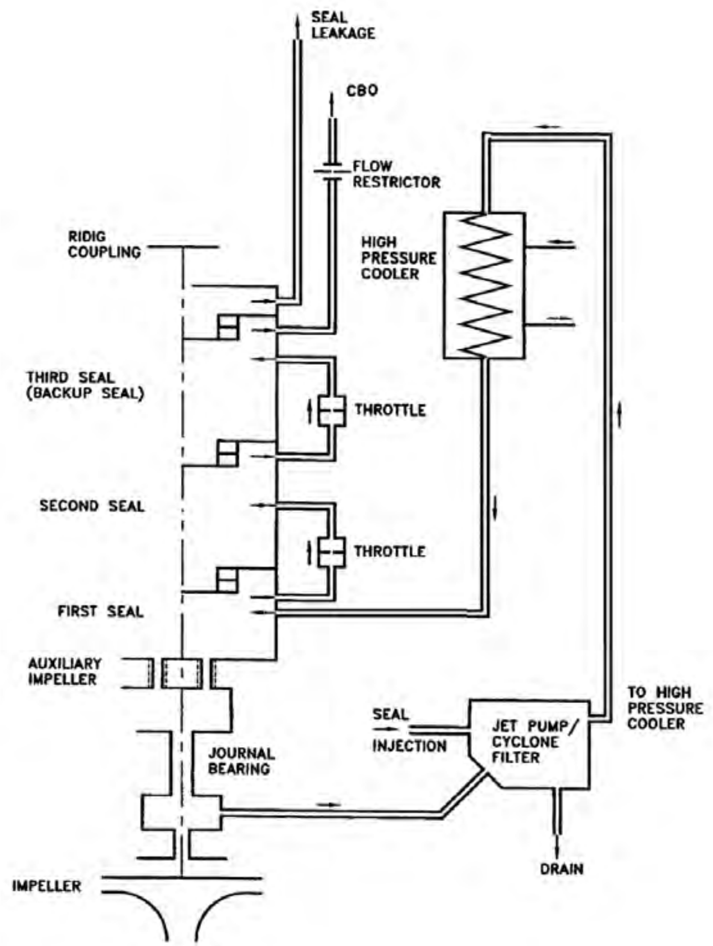
● 5.4.1 Reactor Coolant Pump (2/7)

➢ High Pressure Cooler

- The RCP seals are normally cooled by redundant systems (1) seal injection and (2) heat removal via the high pressure cooler with CCW system

➢ Type F seal

- Materials: Long-term exposure to high temperature and pressure environment
- Test results demonstrated the ability of the seal to survive an extended SBO event.
  - ✓ Simulation of a SBO event lasting up to 72 hours.
  - ✓ Seal performance under low sub-cooling conditions



Reactor Coolant System and Connecting Systems

● 5.4.1 Reactor Coolant Pump (3/7)

➢ **Open Item (RAI No. 8555, Question 05.04-2, Part 1), Date issued (3/15/16); KHNP responded (4/29/16)**

- **Description of Issue** : Clarify regarding RCP seal water injection during 30-minute period (RAI 307-7835, Question 05.04-1, Part 5).
- **Resolution :** RCP seal injection is supplied to the RCPs by two charging pumps or the auxiliary charging pump. Auxiliary charging pump can be powered from the onsite alternate AC power source.

➢ **Open Item (RAI No. 8555, Question 05.04-2, Part 2) , Date issued (3/15/16); KHNP responded (4/29/16)**

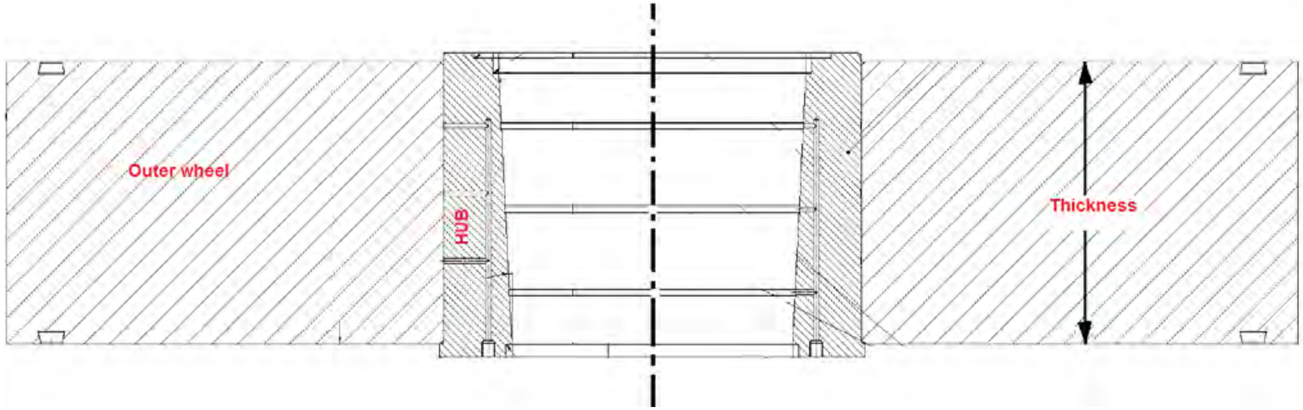
- **Description of Issue** : Commitment regarding the seal test results. (RAI 307-7835, Question 05.04-1, Part 7).
- **Resolution :** Seal test results are available to staff audit. A general summary of tests is incorporated into the DCD.

Reactor Coolant System and Connecting Systems

● 5.4.1 Reactor Coolant Pump (4/7)

➢ Flywheel

- A cylindrical shrink fit of the outer wheel to the hub, and then a conical shrink fit between the hub and the shaft
- Material: 26NiCrMoV14-5, high strength ductile forged material
- Fracture toughness: K_{IC} > 165 MPa√m (150 ksi√in).
- Design: APR1400-A-M-NR-14001-NP, KHNP APR1400 Flywheel Integrity Report



Reactor Coolant System and Connecting Systems

● 5.4.1 Reactor Coolant Pump (5/7)

- **Open Item (RAI No. 8641, Question 05.04.01.01-7), Date issued (7/6/16); KHNP responded (9/12/16)**
  - **Description of Issue** : Describe acceptable approach for determining fracture toughness of the flywheel materials.
  - **Resolution :** Revised DCD to state fracture toughness of at least 150 ksi√in (to specify the direct method of determining fracture toughness)

- **Open Item (RAI No. 8641, Question 05.04.01.01-8), Date issued (7/6/16); KHNP responded (9/12/16)**
  - **Description of Issue** : Clarify operating experience table.
  - **Resolution :** Revised operating experience table with the same material used in Korean and German plants.

Reactor Coolant System and Connecting Systems

● 5.4.1 Reactor Coolant Pump (6/7)

- **Open Item (RAI No. 8641, Question 05.04.01.01-9), Date issued (7/6/16) ; KHNP responded (9/12/16)**
  - **Description of Issue** : Apply a RCP flywheel stress limit of one-third of the yield strength of the material or provide technical justification for use of one-third of ultimate strength.
  - **Resolution :** Apply a RCP flywheel stress limit of one-third of the yield strength of the material according to the SRP 5.4.1.1.

- **Open Item (RAI No. 8641, Question 05.04.01.01-10), Date issued (7/6/16) ; KHNP responded (9/12/16)**
  - **Description of Issue** : Revise the technical report to include an analysis of the hub and provide the appropriate fatigue crack growth rates.
  - **Resolution :** Add a separate stress plot of the hub.

Reactor Coolant System and Connecting Systems

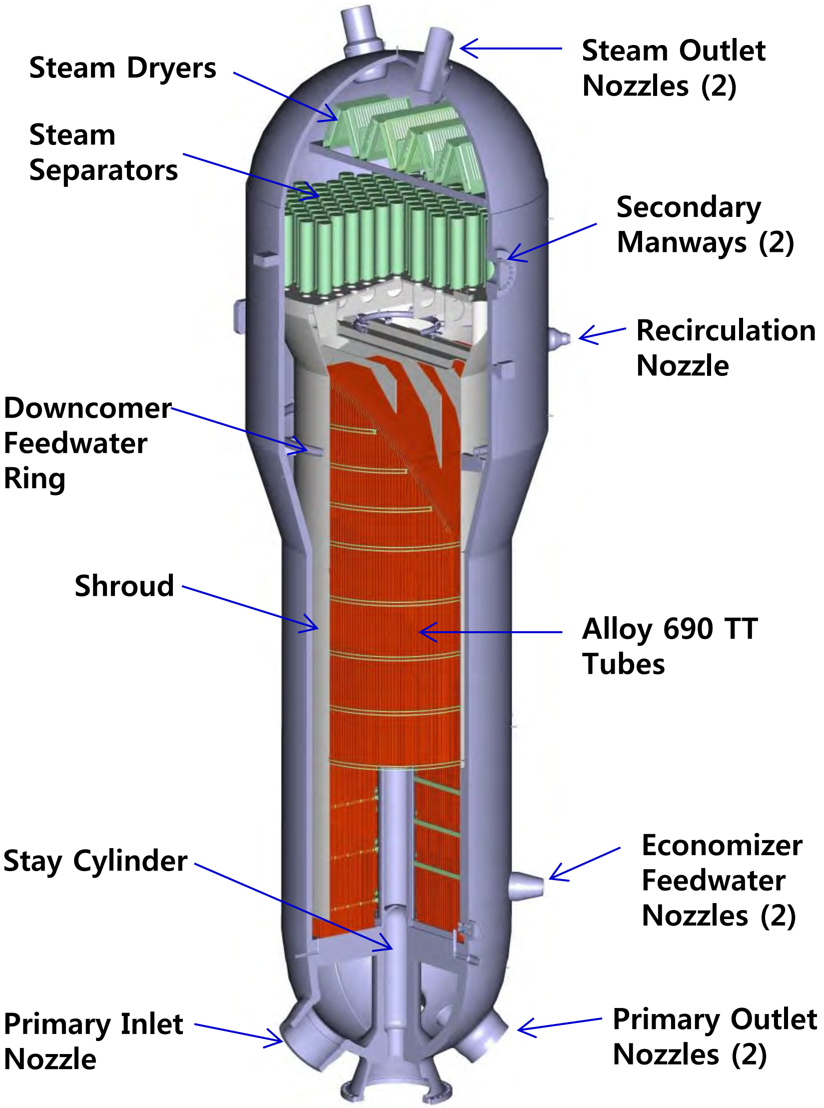
● 5.4.1 Reactor Coolant Pump (7/7)

- **Open Item (RAI No. 8641, Question 05.04.01.01-11), Date issued (7/6/16); KHNP responded (9/12/16)**
  - **Description of Issue** : Specify maximum flaw size used as the acceptance criteria and that it is bounding in determining the critical flaw size.
  - **Resolution :** Revised DCD to include inspection acceptance criteria.
- **Open Item (RAI No. 8641, Question 05.04.01.01-12), Date issued (7/6/16); KHNP responded (9/12/16)**
  - **Description of Issue** : Tests and inspections proposed for the flywheel also apply to the hub.
  - **Resolution :** The hub has oil channels that would make it difficult to perform UT inspection. Flaw propagation will be shown as negligible. Additional defense-in-depth notation will be added to the DCD for ISI instructions.

Reactor Coolant System and Connecting Systems

• 5.4.2 Steam Generator (1/3)

- **Heat Transfer with 2 SGs**
  : 4,000 MWth / 1,400 MWe
- **Alloy 690TT, Vertical U-Tube with 13,102 tubes/SG**
- **Integral Economizer**
- **Steam Outlet Nozzle with Flow Restrictor**



Reactor Coolant System and Connecting Systems

- **5.4.2 Steam Generator (2/3)**

Reactor Coolant System and Connecting Systems

● 5.4.2 Steam Generator (3/3)

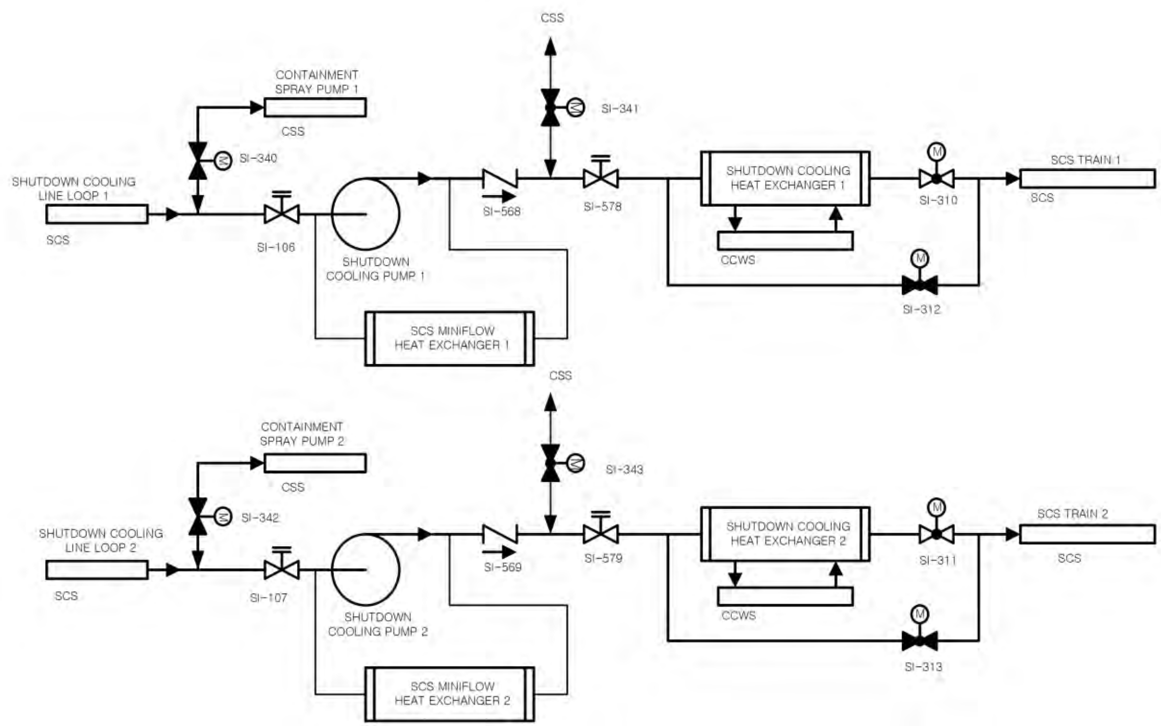
➢ **Open Item (RAI No. 8620, Question 05.04.02.02-6), Date issued (6/8/16); KHNP responded (7/26/16)**

- **Description of Issue** : Administrative issue of the eRAI number and inconsistency of the proposed leakage values between the TS bases and Chapter 15.
- **Resolution**
  - Changed the eRAI number from 05.04.02.02-5 to 05.04.02.02-6
  - Revised the TS bases from 0.3 gpm to 0.6 gpm for all SGs primary to secondary leakage.

Reactor Coolant System and Connecting Systems

● 5.4.7 Shutdown Cooling System (1/2)

- SCS is used to reduce the temperature of RCS from hot shutdown condition (176.7 °C (350 °F)) to refueling condition.
- The SCPs and CSPs are identical and functionally interchangeable.



Reactor Coolant System and Connecting Systems

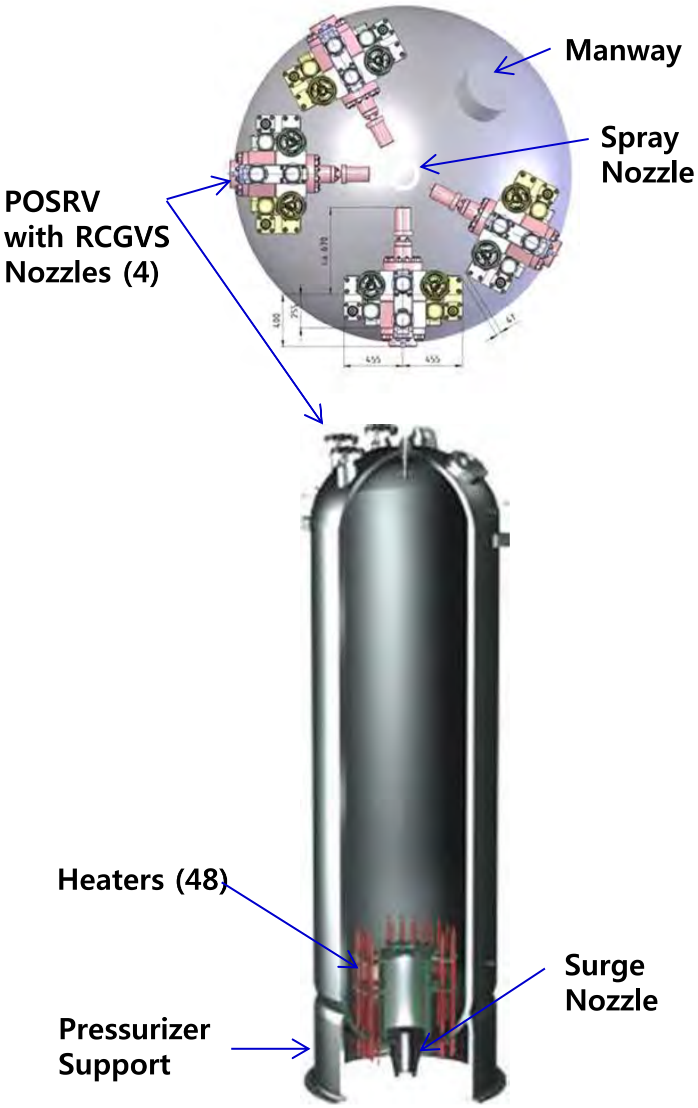
● 5.4.7 Shutdown Cooling System (2/2)

- **Open Item (RAI No. 8614, Question 05.04.07-4), Date issued (5/26/16); KHNP responded (7/8/16)**
  - **Description of Issue** : Revise the ITAAC for gas accumulation addressing GL-2008-1 and NEI-09-10, Revision 1a-A as they relate to SCS, SIS, and CSS.
  - **Resolution :** Revised the ITAAC to include the as-built evaluation for gas entrainment.

Reactor Coolant System and Connecting Systems

• 5.4.10 Pressurizer (1/3)

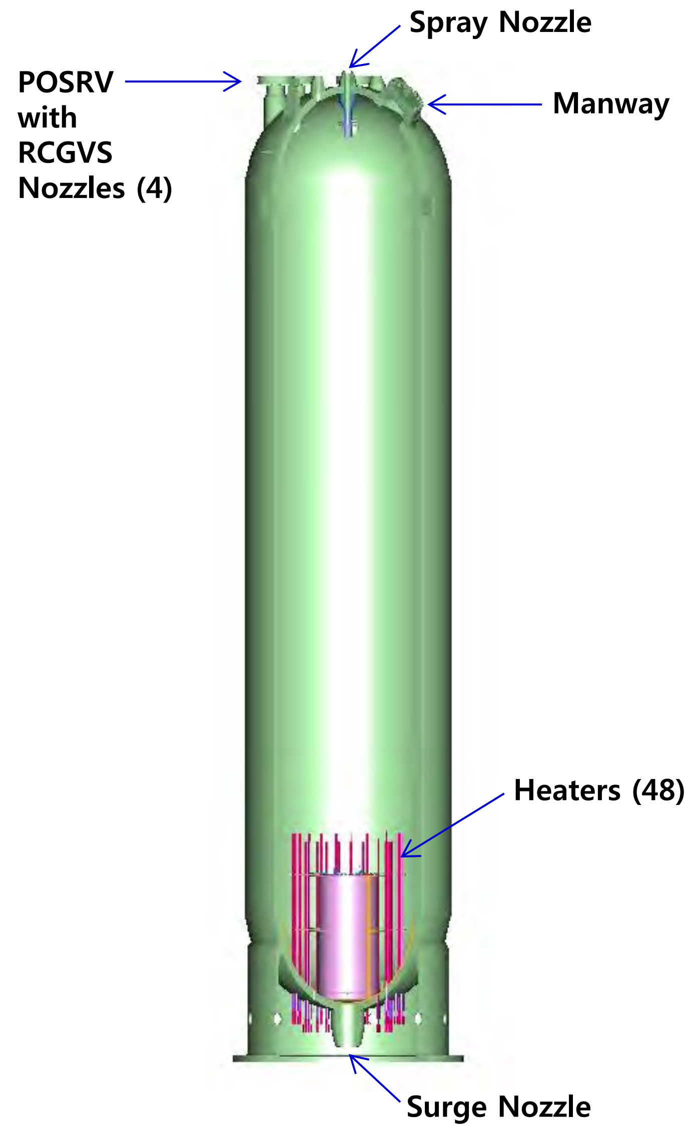
- **RCS Pressure and Volume Control**
  - Internal volume: 2,400 ft^3
  - Connected with surge line
- **4 Pilot Operated Safety Relief Valves (POSRVs)**
  - Overpressurization protection of RCS
  - Integrated installed with reactor coolant gas vent system (RCGVSs)



Reactor Coolant System and Connecting Systems

• 5.4.10 Pressurizer (2/3)

Parameter	Value
Design pressure, kg/cm^{2}A (psia)	175.8 (2,500)
Design temperature, °C (°F)	371.1 (700)
Normal operating pressure, kg/cm^{2}A (psia)	158.2 (2,250)
Normal operating temperature, °C (°F)	344.8 (652.7)
Internal free volume, m^3 (ft^3)	68.0 (2,400)
Normal (full power) operating water volume, m^3 (ft^3)	33.2 (1,171)
Normal (full power) steam volume, m^3 (ft^3)	35.7 (1,260)
Installed heater capacity, kW	2,400
Heater type	Immersion
Spray flow, minimum design capacity, L/min (gpm)	1,703.4 (450)
Bypass spray flow, continuous, L/min (gpm)	1.9~22.7 (0.5~6)
Nozzles:	
Surge, in (nominal) Spray, in (nominal) POSRV, in	12, schedule 160 4, schedule 160 6, Liner ID 7.75, nozzle ID
Instrument:	
Level, in (nominal) Temperature, in (nominal) Pressure, in (nominal)	3/4, schedule 160 1, schedule 160 3/4, schedule 160
Heater, OD, mm (in)	31.75 (1-1/4)



Reactor Coolant System and Connecting Systems

- **5.4.10 Pressurizer (3/3)**

Reactor Coolant System and Connecting Systems

● 5.4.12 Reactor Coolant System High Point Vents

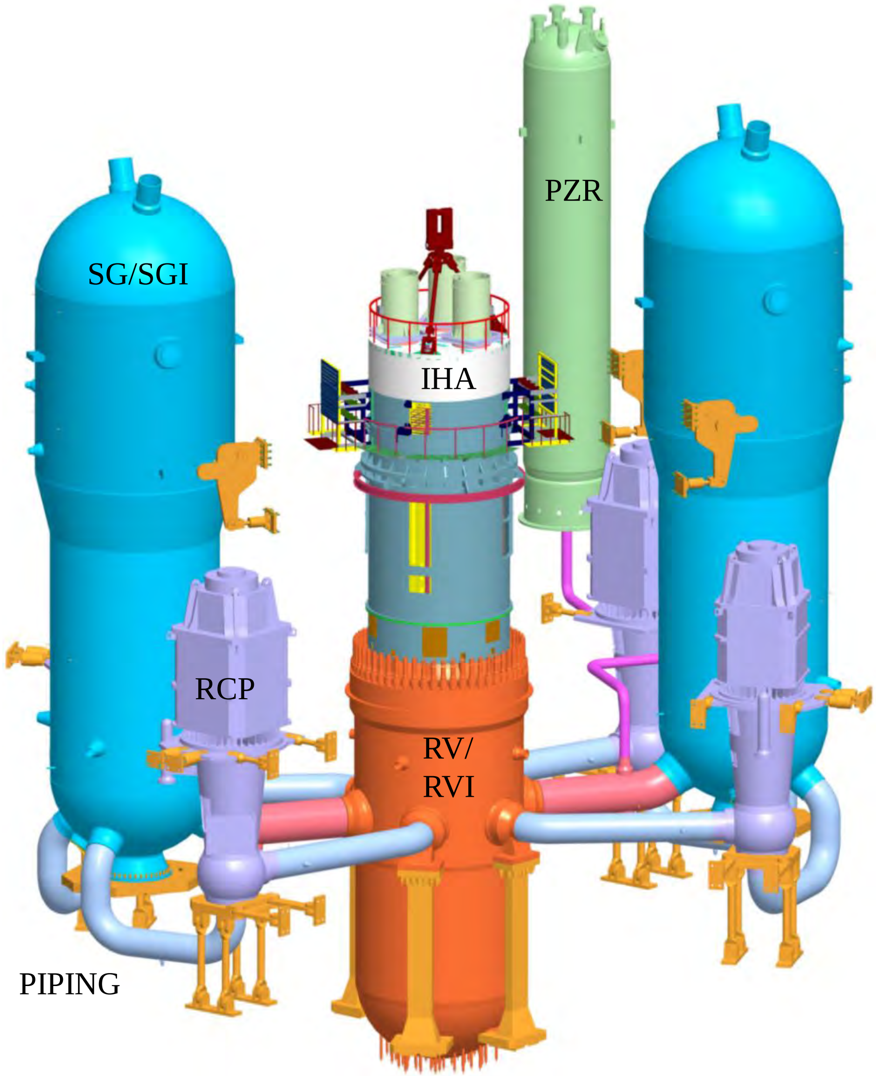
➢ **Reactor Coolant Gas Vent System (RCGVS)**

- Discharge non-condensable gases and steam from the high point of the RCS for venting or pressure control during post-accident conditions
- The in-containment refueling water storage tank (IRWST) collects and condenses the steam discharged from the reactor coolant gas vent system (RCGVS).

Reactor Coolant System and Connecting Systems

• 5.4.15 Component Supports

- **Reactor Vessel**
  - 4 Vertical columns
- **Steam Generator (SG)**
  - 1 Sliding base
  - 2 Snubbers & keys
- **Pressurizer**
  - Integral support
- **Reactor Coolant Pump**
  - 4 Vertical columns
  - 2 Upper & lower horizontal columns
  - 2 Horizontal snubbers



Summary

- In chapter 5, the design features of the RCS of the APR1400 standard plant are described.
- All RCS components are located inside the containment building. Table 5.1.1-1 shows the principal parameters of the RCS.
- The reactor coolant pressure boundary (RCPB) components are consistent with 10 CFR 50.2 and 10 CFR 50.55a. Applicable codes and standards of RCS components are listed in Table 3.2-1

Current Status

- Chapter 5 is on a success path for completion on schedule.
- An SER with Open Items was issued
  - A path for closing each open item has been discussed with the NRC Staff and is being implemented.
  - KHNP believes that the open items except for flywheel which is in progress have been addressed.
- Chapter 5 as reviewed and marked-up in response to NRC's "Request for Additional Information" will be incorporated into the next revision of the Design Control Document.
- Monitoring of changes to other chapters for impacts on this chapter is conducted to maintain consistency.

Attachment : Acronyms

- **ASME: American Society of Mechanical Engineers**
- **CEDM: Control Element Drive Mechanism**
- **CCW: Component Cooling Water**
- **CSP: Containment Spray Pump**
- **CSS: Containment Spray System**
- **EFPY: Effective Full Power Year**
- **ICI: In-Core Instrumentation**
- **IRWST: In-containment Refueling Water Storage Tank**
- **ISI: Inservice Inspection**
- **IST: Inservice Testing**
- **KHNP: Korea Hydro & Nuclear Power**
- **LTOP: Low Temperature Overpressure Protection**
- **POSRV: Pilot-operated Safety Relief Valve**
- **RCGVS: Reactor Coolant Gas Vent System**
- **RCP: Reactor Coolant Pump**
- **RCPB: Reactor Coolant Pressure Boundary**
- **RCS: Reactor Coolant System**
- **RG: Regulatory Guide**
- **RT_{NDT}: Reference Temperature for Nil Ductility Transition**
- **SCS: Shutdown Cooling System**

NON-PROPRIETARY

Attachment : List of COL Items for Ch. 5 (1/3)

COL No.	Description
COL 5.2(1)	The COL applicant is to address the addition of ASME Code Cases that are approved in NRC RG 1.84 at the time of the application.
COL 5.2(2)	The COL applicant is to address the ASME Code Cases approved in NRC RG 1.147 at the time of the application and invoked for the ISI program of a specific plant.
COL 5.2(3)	The COL applicant is to address the ASME Code Cases approved in NRC RG 1.192 at the time of the application and invoked for operation and maintenance activities of a specific plant.
COL 5.2(4)	The COL applicant is to address the material specifications, which are not shown in Table 5.2-2, as necessary.
COL 5.2(5)	The COL applicant is to specify the version of EPRI's, "Primary Water Chemistry Guidelines," that will be implemented.
COL 5.2(6)	The COL applicant is to address the actual, as-procured, fracture toughness data of the RCPB materials to the staff at a predetermined time by an appropriate method.
COL 5.2(7)	The COL applicant is to submit the actual, as-procured yield strength of the austenitic stainless steel materials used in RCPB to the staff at a predetermined time agreed-upon by the regulatory body.
COL 5.2(8)	The COL applicant is to provide and develop the implementation milestones for the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.
COL 5.2(9)	The COL applicant is to address the provisions to accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design.

Attachment : List of COL Items for Ch. 5 (2/3)

COL No.	Description
COL 5.2(10)	The COL applicant is to provide the list of Code exemptions in the ISI program of the specific plants, if it exists.
COL 5.2(11)	(Deleted)
COL 5.2(12)	(Deleted)
COL 5.2(13)	The COL applicant is to prepare and implement a boric acid corrosion (BAC) prevention program in conformance with Generic Letter 88-05.
COL 5.2(14)	The COL applicant is to prepare the preservice inspection and testing program.
COL 5.2(15)	The COL applicant is to address and develop the milestones for the preparation and implementation of the procedure for operator responses to prolonged low-level leakage.
COL 5.2(16)	The COL applicant is to address the portions of the later editions and addenda of ASME Code.
COL 5.3(1)	The COL applicant is to provide a reactor vessel material surveillance program for a specific plant.
COL 5.3(2)	The COL applicant is to develop P-T Limit curves based on plant-specific data.
COL 5.3(3)	The COL applicant is to verify the RTPTS value and the USE at EOL based on plant-specific material property and neutron fluences.

Attachment : List of COL Items for Ch. 5 (3/3)

COL No.	Description
COL 5.3(4)	The COL applicant is to provide and develop the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.
COL 5.4(1)	The COL applicant is to prepare operational procedures and maintenance programs related to leak detection and contamination control of RCS.
COL 5.4(2)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of RCS.
COL 5.4(3)	The COL applicant is to prepare operational procedures and maintenance programs related to leak detection and contamination control of SCS.
COL 5.4(4)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of SCS.
COL 5.4(5)	The COL applicant is to verify the as-built RV support material properties and 60-year neutron fluence.
COL 5.4(6)	The COL applicant is to prepare a PSI and ISI programs of the SG tubes.
COL 5.4(6)	The COL applicant is to provide a procedure to protect the plant personnel from harsh environment and lengthen the work time within containment during hatch closure.
COL 5.4(7)	The COL applicant is to prepare an inspection and monitoring program for the cladding material integrity of SG channel heads.

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Korea Hydro Nuclear Power Co., Ltd (KHNP)

APR1400 Design Certification Application Review

Safety Evaluation with Open Items: Chapter 5

REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

SEPTEMBER 22, 2016

- **Technical Staff Presenters**
  - John Budzynski – DCD Sections 5.2.2, 5.4.1.2, 5.4.7, & 5.4.1.12
  - Dan Widrevitz – DCD Section 5.2.3
  - Jessica Umaña – DCD Section 5.2.5
  - John Honcharik – DCD Section 5.4.1.1
  - Greg Makar – DCD Sections 5.4.2.2

- **Project Managers**
  - Jeff Ciocco – Lead Project Manager
  - Jessica Umaña – Chapter 5 Project Manager

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  Materials & Chemical Engineering Branch
- **Steven Downey**
  Materials & Chemical Engineering Branch
- **Eric Reichelt**
  Materials & Chemical Engineering Branch

Overview of Design Certification Application, Chapter 5

SRP Section/Application Section		No. of Questions	Number of OI
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 5055a	3	0
5.2.1.2	Compliance with Applicable Code Cases	2	0
5.2.2	Overpressure Protection	7	3
5.2.3	Reactor Coolant Pressure Boundary Materials	22	1
5.2.4	Inservice Inspection and Testing of the Reactor Coolant Pressure	2	0
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	3	0

Overview of Design Certification Application, Chapter 5

SRP Section/Application Section		No. of Questions	Number of OI
5.3.1	Reactor Vessel Materials	0	0
5.3.2	Pressure-Temperature Limits, Pressurized Thermal Shock, and Upper-Shelf Energy Data and Analyses	1	0
5.3.3	Reactor Vessel Integrity	0	0
5.4	Reactor Coolant System Component and Subsystem Design	2	1
5.4.1.1	Reactor Coolant Pump Flywheel Integrity	12	6
5.4.1.2	Reactor Coolant Pump	0	0

Overview of Design Certification Application, Chapter 5

SRP Section/Application Section		No. of Questions	Number of OI
5.4.2.1	Steam Generator Materials	3	0
5.4.2.2	Steam Generator Program	6	1
5.4.3	Reactor Coolant Piping	0	0
5.4.7	Residual Heat Removal System	4	1
5.4.10	Pressurizer	0	0

Overview of Design Certification Application, Chapter 5

SRP Section/Application Section		No. of Questions	Number of OI
5.4.11	Pressurizer Relief Tank	0	0
5.4.12	Reactor Coolant System High-Point Vents	10	0
Totals		77	13

Technical Topics

- System Design
  - Overpressure protection for (1) RCS, (2) primary side of systems connected to RCS, and (3) secondary side of SGs

    - RCPB overpressure protection consist of 4 POSRV, 2 SCS RV, and 20 MSSV (5 per steamline)
  - SCS suction line relief valves provide sufficient pressure relief capacity to mitigate the most limiting low temperature overpressure protection (LTOP)

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Open Items

- **RAI 8244, Question 05.02.02-1** – Provide information regarding the analysis performed to demonstrate that POSRV and SCS RV provide sufficient capacity to maintain the pressure below the limit during full power and LTOP conditions.
  - **Status:** The response was received on 12/11/2015. The POSRV analysis was adequately addressed; however, the LTOP analysis did not address staff concerns related to methodology, codes, input parameters, and assumptions used in the LTOP analysis. This question is closed unresolved.
- **RAI 8609, Question 05.02.02-7 Follow up question** - Provide a description of the analysis methodology, computer codes, input parameters, and assumptions used in the analysis of the limiting LTOP event(s). Provide justification that the inputs are suitably conservative.
  - **Status**: The response was received on 6/9/2016. The follow up RAI response is currently in evaluation.
- **RAI 8244, Question 05.02.02-5 -** Address the difference between the pre-service testing requirements and Chapter 14 test.
  - **Status**: The response was received on 6/9/2016. The response was found acceptable because the complete pre-service testing is included in the Section 14.2 rewrite. The staff confirmed the revision. This question is closed resolved.

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Technical Topics

- Pump Design
  - RCP is a vertical shaft, single-stage, centrifugal pump
  - Loss of Component Cooling Water Event

    - CCW flows to pump/motor bearing oil coolers, motor air coolers, and high-pressure seal water cooler stopped

    - CVCS continues to provide seal water injection flow to the RCP seals

    - Pump/motor bearings designed to withstand loss of CCW for 30 minutes

    - CCWS Division 1 supports charging pump mini flow heat-exchanger in CVCS and all coolers on the RCPs
  - RCP seal design is improved to have low seal leakage rates during a station black-out (SBO) event

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Open Items

- **RAI 8555, Question 05.04-2, Part 1** – With respect to Loss of Component Cooling Water, provide a response to clarify whether the loss of CCW Division 1 affects the charging pump operation during the 30-minute period following the event
  - **Status:** The response was received 4/29/2016. The response was found acceptable because of the loss of the CCW to the charging pump only has a long term effect on its operation. This question is closed resolved.
- **RAI 8555, Question 05.04-2, Part 2** – Revise DCD Section 5.4 to include a summary of the seal leakage test results that show the low seal leakage rate during a simulated SBO event.
  - **Status:** The response was received on 4/29/2016). The response was found acceptable because the DCD is being revised to include a reference to the seal test report. This question is closed resolved.

Technical Topics
Section 5.4.7 – Shutdown Cooling

Technical Topics

- System Design
  - Configuration Similar to Current PWR Designs,
    - Two independent trains – SCP interchangeable with CSP
  - Used during startup/shutdown and refueling operations
    - During shutdown and refueling RHR is placed in service below approximately 450 psig and 350°F
- Evaluated against GL-2008-01 and RIS 2013-09
- Evaluated against NEI 09-10 Rev 1a-A – Prevention and Management of System Gas Accumulation

Technical Topics

Section 5.4.7 – Shutdown Cooling

Open Items

- **RAI 8614, Question 05.04.07-4** – Provide ITAAC to the SCS design that addresses GL-2008-01 and NEI 09-10 with respect to managing gas accumulation during plant operations
  - **Status:** The response was received on 7/8/2016. The ITAAC was adequately addressed and in compliance with NEI 09-10 Revision 1a-A; however, NEI 09-10 was referenced in the DCD. The staff will discuss the need to include the reference with the applicant. This question currently remains in evaluation.

Technical Topics

- 10 CFR 50.34(f)(2)(vi) and TMI Action Plan Item II.B.1 require high point vents, and 10 CFR 50.46a specifies acceptance criteria
  - Remove non-condensable gases from high points in the RCS to maintain adequate core cooling (enhance natural circulation)
  - The RCS high points are the reactor vessel head, pressurizer, and steam generator U-tubes; however, individual U-tubes are not required to have high point vents (10 CFR 50.46a)
- APR1400 High Point Vent Design: Standard Configuration
  - Safety-grade vent paths: (1) reactor vessel head to in-containment refueling water storage tank (IRWST); and (2) pressurizer head to IRWST
  - Each safety-grade vent path has two parallel vent paths with two normally closed solenoid-operated valves in series
  - Small-break loss-of-coolant accident analyses bound potential vent line breaks upstream of isolation valves
- Operating procedures described at a high level in the DCD; detailed operating procedures to be developed by COL applicant

Open Item

- **Open Item, DCD Clarification** - It was determined that a portion of DCD text providing additional detail beyond the staff criteria for establishing that austenitic materials used for the RCPB will not be sensitized should be removed from the DCD. Since the DCD will be granted finality by rulemaking, this removal is necessary as the staff could not make a specific regulatory finding regarding its accuracy.
    - **Status:** The applicant has been notified and the staff has scheduled a meeting with the applicant to review this.

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Technical Topics

- Regulatory Requirements: GDC30, GDC2

- Review Guidance: SRP Section 5.2.5
  For GDC 30 on RCPB leakage detection, the review is based on RG 1.45.
  For GDC 2 on seismic design, the review is based on RG 1.29 and RG 1.45.

- Review Areas: leakage detection capability, sensitivity, and response time; leakage detection systems; seismic qualification; leakage instrumentation in the control room; prolonged low-level RCS leakage; separation of identified and unidentified leakage; intersystem leakage; plant TSs; initial testing program; ITAAC; and COL information items.

- Three RAIs were issued and all are being resolved pending confirmation in the next revision of the DCD.

- One COL information item.

Technical Topics

Section 5.4.1.1 – Reactor Coolant Pump (RCP) Flywheel Integrity

Technical Topics

RCP flywheel analyzed to prevent fracture and possible missile.

- KHNP provided:
  - Material selection, fabrication techniques, preservice and inservice inspections and overspeed testing per NUREG-0800
  - Analysis per RG 1.14 (APR1400-A-M-NR-14001)
    - Critical speeds for ductile and non-ductile fracture
    - Fatigue crack growth

Open Items

- **RAI 8641, Question 05.04.04.01-7** – The applicant needs to provide the method for determining fracture toughness.
  - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-8** - The applicant needs to provide operating experience for the flywheel material 26 NiCrMoV 14-5.
  - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-9** - The applicant needs to provide analysis using design acceptance criteria of 1/3 yield strength or provide justification for using 1/3 ultimate strength as the design acceptance criteria.
  - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-10** - The applicant needs to provide analysis of the hub and why crack growth rates used for the flywheel are applicable to the specific flywheel material.
  - **Status:** This question is currently waiting a response.
- **RAI 8641, Question 05.04.01.01-11** – The applicant needs to provide in the DCD the maximum flaw size used in the analysis and that can be detected by the applicable NDE method.
  - **Status:** This question is currently waiting a response.

Technical Topics
Section 5.4.1.1 – Reactor Coolant Pump (RCP) Flywheel Integrity

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Open Items (continued)

- **RAI 8641, Question 05.04.01.01-12** - The applicant needs to include in the DCD the applicable PSI and ISI for the hub that is consistent with the flywheel to ensure the integrity of the hub is maintained.
    - **Status:** This question is currently waiting a response.

Open Items

- **RAI 494-8620, Question 05.04.02.02-6** - The assumed values for primary-to-secondary leakage described in the TS Bases (B 3.4.12 and B 3.4.17) appeared to be inconsistent with the accident analyses described in Chapter 15. Question 05.04.02.02-6 requested clarification of the leakage values in the TS Bases and how they are consistent with applicant's accident analyses and the Standard Technical Specifications (STS). Consistency between the accident analyses, APR1400 TS, and STS is needed to establish and implement a Steam Generator Program to ensure that tube integrity is maintained. This is part of meeting GDC 32.
  - **Status:** The response was received 7/26/2016. The response clarified the accident analysis assumptions and proposed revisions to TS Bases Sections B 3.4.12 and B 3.4.17 that are consistent with the accident analyses and STS. Therefore, the response is acceptable, this open item will be **closed,** and Question 5.04.02.02-6 will be tracked as a confirmatory item.

ACRONYMS

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10 CFR – Title 10 of the Code of Federal Regulations
CCW – component cooling water
CVCS – chemical and volume control system
COL – combined license
DV – depressurization valve
FSAR – Final Safety Analysis Report
MCR – main control room
MOV – motor operated valve
PRT- pressurizer relief tank
PTLR – Pressure – Temperature Limits Report
PTS – pressurized thermal shock
PWR – pressurized water reactor
RAI – request for additional information
RCP – reactor coolant pump
RCPB – reactor coolant pressure boundary
RCS – reactor coolant system
RCSHPV – reactor coolant system high point vent
RG – Regulatory Guide
RHR – residual heat removal
RVHVS – reactor vessel head vent system
RWSP – refueling water storage pit
SDV – safety depressurization valve
SER – safety evaluation report
SRP – Standard Review Plan
SRV – safety relief valve