APR1400 Probabilistic Risk Assessment (PRA), Severe Accident (SA), and Reliability Assurance Program (RAP) - Topics for Discussion (03/26/2015)					
The following questions are not requests for additional information (RAIs), but are for discussion only					
Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
Sections 19.0 and 19.1 "PRA"					
Technical issue - Issue that may impact the technical contents of the submittal and staff's ability to reach a reasonable assurance finding.					
PRA-1	Technical issue	Section 19.1, "Probabilistic Risk Assessment," does not address asymmetric configuration and modeling. It is not clear whether the APR1400 PRA model is symmetric or not.	Hanh Phan	The modeling is not symmetric with respect to operating systems. Operating systems, like component cooling water, service water, etc. have assumed configurations. In addition, the AAC gas turbine, which can only support two of four 4kV SWGR on a single division is assumed to be aligned to the 1A and 1C SWGR. These configurations are documented in the system notebooks.	Young In
PRA-2	Technical issue	Section 19.1, "Probabilistic Risk Assessment," validation on the modeling of digital I&C including its adequacy, completeness, and common cause failures could not be found. The staff found no risk significance regarding digital I&C.	Courtney St. Peters	Digital I&C systems are documented where EF system, APR1400-K-P-NR-013217-P and RP system, APR1400-K-P-NR-013218-P. The digital I&C software reliability study has not been evaulated yet.	Young In
PRA-3	Technical issue	Section 19.1.5, "Safety Insights from the External Events," in accordance with SRP Chapter 19 guidance, the staff could not find any analyses for the applicable external hazards, especially since deterministic evaluations were addressed.	Hanh Phan	Other external evaluation notebook (APR1400-K-P-NR-013801-P) is available, which can be made available in the electronic reading room.	Ray Dremel
PRA-4	Technical issue	Section 19.1.6, "Safety Insights from the PRA for Other Modes of Operation," DCD Section 5.4.7 does not provide enough information for the staff to conclude that air entrainment during reduced inventory conditions has been adequately addressed deterministically, which impacts the estimated reduced inventory risk.	Marie Pohida	The Shutdown Evaluation Report (APR1400-E-N-NR-14005-P) including prevention of air entrainment during reduced inventory operation has been submitted and Section 2.3 provides design features regarding prevention of air entrainment in APR1400 design. For more enough information refer to Section 2.3 in the Shutdown Evaluation Report.	KEPCO E&C SD
PRA-5	Technical issue	Section 19.1.6 , appears to have omitted initiating events (such as RCS overdraining- basic event %SO) from the risk achievement analyses.	Marie Pohida	Risk importance for individual initiating events, for all shutdown states, are provided in the tab entitled 'IE Importance.' These results are integrated over all Plant Operating States. LPSD Internal Events Level 1 Quantification Notebook (APR1400-K-P-NR-013707-P, R1) included the results from the risk achievement analyses except for initiating events. The future revision of notebook can include the results from the risk achievement analyses for initiating events, if necessary.	Ross Anderson Kim Jae Gup
PRA-6	Technical issue	Section 19.3.2.3, "Mid-Loop Operation," regarding station blackout mitigation strategies, gravity feed from the SITS is utilized to prevent core uncovery. Given a high elevation vent in the RCS, such as an open pressurizer manway, gravity feed may not be feasible due to surgeline flooding in the pressurizer. The staff could find no evaluations on the station blackout strategies given an open RCS with a high elevation vent.	Marie Pohida	The evaluation on the station blackout strategy for mid-loop operation is provided in Section, A.5.3 in Appendix A of APR1400-E-P-NR-14005-P, "Evaluations and Design Enhancements to Incorporate Lessons Learned from the Fukushima Dai-Ichi Nuclear Accident." The evaluation results show that core uncovery can be prevented until more than three hours by utilizing gravity feed from two SITs, while the vapor generated from the core is vented through the pressurizer manway. In this analysis, it is assumed that the operators depressurize SIT, but still maintain slightly pressurized to overcome flooding resistance of surge line and pressurizer.	KEPCO E&C SD Ross Anderson

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-7	Technical issue	SRP Section 14.3 "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)," indicates that "the important insights and assumptions from the PRA provided in FSAR Chapter 19 should be used to determine the appropriate top-level design features for inclusion in Tier 1. A discussion of how the important insights or assumptions from the PRA should be addressed in the selection of the Tier 1 material. The important integrated plant safety analyses from Tier 2 should be considered, such as analyses of internal events, fires, floods, severe accidents, and shutdown risk." Accordingly, it is unclear how the APR1400 PRA was used in determining the scope of ITAAC and which ITAAC that were derived from the important	Hanh Phan	Will review the risk insights and assumptions to ensure that the latest information are incorporated.	Young In
PRA-8	Technical issue	Page 19.1-260, "Key Assumptions," No. 5, "Room cooling is assumed not to be needed for the following rooms. The room heatup calculations to be supplied are expected to show that room cooling is not required. This assumption applies to each room when both the emergency HVAC and ECW are lost." The staff could not find the room heatup calculations for review. It is unclear why HVAC is not modeled as an IE.	Hanh Phan	The room heatup calculations were performed only for those SSCs related to the Fukushima actions only (e.g., ACP, TDP, SIP, associated SSCs). The room heatup calculations for the rooms specified in Assumption 5 in Table 19.1-4 will be provided within the schedule to be established.	Young In
PRA-9	Technical issue	Page 19.1-110 states "This CCFP is the conditional probability of a large release (CPLR) for operations at power." The staff could not find the basis for assuming the conditional containment failure probability is equal to the conditional probability of a large release.	Jason Schaperow	The wording of the sentence is confusing as written. It states that the CCFP presented as the risk metric is the CCFP of large releases. The statement about "This CCFP …" is therefore referring to the definition in that paragraph, which is really a definition of CPLR and not the traditional definition of CCFP. The paragraph will be rewritten as "The Conditional Probability of Large Release (CPLR) from all internal events (at power) is 8.4 x 10-2." The statement about the NRC goal for CCFP is removed since SECY-10-0121 states that there is no design objective for CCFP for advanced reactors.	Jeff Leary
PRA-10	Technical issue	An RCP seal leakage and failure model is needed to estimate severe accident progression and source term. The following statements related to the RCP seal leakage and failure model are made in Chapter 19: "RCP seal LOCA probability, given a total loss of seal cooling and the RCP trip, is assumed to be equal to 1x10-3 per pump." "A detailed seal LOCA model will be developed when the RCP seal LOCA technical bases, including the seal LOCA probabilities, become available." When will a detailed seal LOCA model become available? The staff could not find the seal leakage and failure model (including seal leak sizes and/or flow rates) assumed for the APR1400 PRA, including the basis for the model.	Jason Schaperow	The RCP seal test will be done in the fall of this year, and the RCP seal probability calculation will take 2-3 months afterward. The PRA modeling scope or approach will be decided afterward.	Young In
PRA-11	Technical issue	Section 19.1.4.1.1.1, it is not clear if there are any recovery actions considered during the IE analysis, as mentioned in ASME/ANS PRA Standard IE-C11.	Ayo Ayegbusi	In general, there is no credit given for recovery actions in the initiating events analysis except for the support system initiating events (SSIEs), as documented in APR1400-K-P-NR-013101.	Steven Phillippi
PRA-12	Technical issue	Section 19.1.4.1.1.1, it is not clear if there is a ISLOCA analysis/calculation performed going by note 5 of Table 19.1-6, as mentioned in ASME/ANS PRA Standard IE-C14.	Ayo Ayegbusi	Table 19.1-6 Note (5) states that the ISLOCA initiating event frequency (/rcy) is taken from calculation. There is an actual ISLOCA calculation for the APR1400 design that determined the ISLOCA initiating event frequency, Attachment A of APR1400-K-P-NR-013101.	Steven Phillippi

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-13	Technical issue	To verify the accident sequence analysis in Section 19.1.4.1.1.3 meets the requirements of ASME/ANS PRA Standard SC-A5, it is not clear whether any sequences that exceeded the 24 hour mission time to achieve a stable condition and what assumptions were made for those sequences.	Ayo Ayegbusi	All sequences that were considered success were safe and stable for periods in excess of 24 hours. No assumption were needed or used to justify success for cases where core damage was prevented for 24 hours but actions were needed in the near future to prevent core damage. See Success Criteria Notebook (APR1400-K-P-NR-013103-P).	Ray Dremel
PRA-14	Technical issue	In DCD Section 19.1.6.2.2.5, "Key Assumptions," the DCD states, " B. Failure of hydrogen control from PARs and/or igniters is assumed to yield a conditional probability of containment rupture due to hydrogen detonation of 0.1, plus another conditional probability of containment rupture due to hydrogen burn of 0.1 or 0.01. These probabilities are believed to be conservative, but additional calculations are needed for confirmation." The staff could not find the results of the calculations documenting the conditional containment failure probability due to hydrogen.	Marie Pohida / Jason Schaperow	The hydrogen calculations to support the assumptions were not complete at the time of the DCD, so this key assumption was made to bound the effect of hydrogen-induced containment failure. The hydrogen calculation has been performed to verify successful hydrogen control during LPSD. The analyses consider the following conditions. (1) RCS Intact or Not Intact – Includes hydrogen release locations such as Steam Generator Inlet Plenum manway and Pressurizer manway, when the RCS is open; (2) The potential unavailability of hydrogen mitigation features during planned outage - A detailed conditional probability of containment rupture due to hydrogen burn has not been performed, but the calculations demonstrate that the probabilities assumed are conservative. The results confirm that the assumptions made were conservative, and the updated analysis will either retain the conservative assumptions or refine the numbers to be more realistic. In either case, the revised analysis will present the results of the detailed hydrogen calculations.	Jeff Leary
<u>Verification / justification</u> - Additional information may be necessary for the staff to evaluate conformance with the SRP or technical guidance and reach a reasonable assurance finding.					
PRA-15	Verification / justification	Section 19.1.2, "Quality of PRA," additional justification of the "Gap Analysis" and the technical adequacy of internal fires PRA and PRA-based SMA are not found.	Hanh Phan	The gap analysis has not been performed. A self-assessment for the non-Peer Reviewed elements will be conducted and the results presented in Table 19.1-1 (similar to what was done for internal events).	Joe Lavelline
PRA-16	Verification / justification	Section 19.1.2, "Quality of PRA," the applicant referenced the requirements in ASME/ANS 2009 PRA Standards for its "Gap Analysis," however the previous PRA Standards nomenclature was used for supporting requirements when reporting the results in the DCD.	Hanh Phan	It is not clear what specific "nomenclature" is being referred to in the comment, this should be clarified.	Joe Lavelline
PRA-17	Verification / justification	Section 19.1.4.2, "Level 2 Internal Events PRA for Operations at Power," descriptions of the severe accident physical processes/phenomena and the success criteria used to delineate accident sequences are not clear.	Hanry Wagage	The severe accident physical processes and phenomena are presented in pages 19.1-69 through 19.1-82 for PDSs, and 19.1-84 through 19.1-87 and also 19.1-89 through 19.1-102 for CETs. To ensure the APR1400 DC submittal can address this item, please identify if there are specific areas within these ranges that are not clear.	Jeff Leary
PRA-18	Verification / justification	Section 19.1.6, appears to screen risk from a water solid condition given that the SCS relief valves are used for Low Temperature Overpressure Protection (LTOP). If these SCS relief valves are challenged but fail to reseat, there could be a low elevation RCS leak path as opposed to a high elevation leak path from the pressurizer. The staff could not find justification regarding the screening of shutdown events occurring in a water solid condition.	Marie Pohida	Plant Operating State 3B is the cooldown below 212°F on the Shutdown Cooling system, which is typically a water-solid evolution. The analyses consider sequences in which decay heat removal is lost and the LTOP valves fail to reclose. LTOP valves are in-service at Mode 4, 5, and 6. LTOP valves can be opened when RCS is Intact and RCS is pressurized after loss of SCS cooling. The analyses for the POS3A, 3B, and 13 consider sequences in which decay heat removal is lost and the LTOP valves fail to reclose. Notebook APR1400-K-P-NR-013702-P Appendix A includes POS3A, 3B, and 13 Event Trees.	Ross Anderson Kim Jae-Gup

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PRA-19	Verification / justification	DCD Tier 1, Page 1.2-1 states: "The design target for CDF is 1E-5 events per reactor year, and the design target for LRF is 1E-6 events per reactor year. These targets include an assessment of internal and external events, excluding seismic events, sabotage, and other external events, and an assessment of shutdown events." However, in Chapter 19, the total CDF and LRF are estimated to be 7.7E-6/yr and 5.6E-7/yr, respectively, without seismic and other external events contributions and these design targets could be exceeded when including seismic risk and other external events risk.	Hanh Phan	The PRA-based SMA methodology used satisfies the recommendation of SECY-93-087 (Reference 8) approved by the NRC for a seismic risk evaluation. While quantitative results are not produced, this analysis ensures that the plant design is sufficiently robust to meet the design targets CDF and LERF. COL19.1(7) requires that the COL applicant "confirm that the PRA-based seismic margin assessment is bounding for the selected site, and to update the assessment to include site-specific SSC and soil effects (including sliding, overturning liquefaction, and slope failure). The COL applicant is to confirm that the as-built plant has adequate seismic margin." The "other external" events were subjected to a screening evaluation presented in Section 19.1.5.4 per the process used in the ASME/ANS Standard. This process, when applied in the design phase and COL phases ensures that the plant design is sufficiently robust to meet the design targets CDF and LERF. It should be noted that the susceptibility to other external hazards is highly site-specific. COL item 19.1(8) requires that the applicant address the non-screened events. A COL action item needs to be added to	Joe Lavelline
PRA-20	Verification / justification	Tier 1 Page 1.9-99, the term "Not applicable (COL)" is not defined.	Hanh Phan	Replace this wording with "This item is not applicable to the design certification phase since it requires site specific information. This item will be addressed in the COL phase." Note that we must ensure that there is a COL action item that addresses the requirement.	Joe Lavelline
PRA-21	Verification / justification	Page 19.1-2 states that "If sufficient information is not available, then the information from the reference plants is used. The reference plants are Shin-Kori Units 3 and 4." Since Shin-Kori Units 3 and 4 are under construction, it is unclear what/how the information from these plants has been used.	Hanh Phan	Specific use of reference plant information is contained within the PRA notebooks. In addition, the DCD Tier 2 document contains some information. For example, in Section 19.1.4.1.1.3.f, it stats that the bases for features and operating procedures are the APR1400 EOGs, but additional bases are from the reference plants. Section 19.1.4.1.2.5.h. notes that the digital I&C system model retains the current hardware model from the reference plant, except for the software events and the communication link models. Section 19.1.5.1.1.e notes that where available, information from the reference plant is used for the component fragilities (also see Table 19.1-43). Section 19.1.6.3.1.2.ee. (page 19.2-201) states that the location of the offsite power cables in the turbine building are based on the reference plant. Table 19.1-14 identifies equipment failure modes based on reference plant data.	Greg Rozga / Ray Dremel Kim Sung Hyun
PRA-22	Verification / justification	DCD Chapter 1, Page 1.9-76, states that APR1400 conforms to SRP Section 19.1 "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests after Initial Fuel Load" however, SRP Section 19.1 is only applicable to the amendment requests after initial fuel load.	Hanh Phan	This should be non-applicable. DCD will be revised.	Joe Lavelline
PRA-23	Verification / justification	DCD Page 19.1-378 includes the basic event, "RC-POSRV V200/201/202/203." The staff is unsure whether this represents a CCF event for the POSRVs and if so, whether it was modeled in models other than the Level 1 internal model.	Tony Nakanishi	RC-POSRV V200/201/202/203 is not a CCF event, but refers to a representation of SSCs with same importance values for similar SSCs in different trains. Note that the indetifiers in Table 19.1-20 are equipment IDs, not basic events.	Joe Lavelline

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PRA-24	Verification / justification	Figure 19.1-49 suggests that containment failure is avoided in 86.2% of severe accidents. What is causing containment failure to be avoided in these cases? What is preventing late overpressure failure of the containment in these cases?	Jason Schaperow	The intact containment states have success of either the containment spray or Emergency Containment Spray Backup System (ECSBS), which provide the long term containment pressure control. The two dominant contributors to the intact containment state are: 1) The reactor fails but does not induce containment failure, the cavity has a significant volume of water on it, and containment heat removal is successful, and 2) The reactor integrity is maintained by in-vessel retention, vessel failure does not occur, and containment heat removal is successful.	Jeff Leary
PRA-25	Verification / justification	One of the surrogate safety goals is that the large release frequency should be less than 1E-6/year. Should this number (1E-6) be compared with the PRA result of the total large release frequency for all initiators and operating modes? What is the resulting total large release frequency for all initiators for APR1400?	Jason Schaperow	The total LRF from all modes and contributors should be summed. The LRF for at-power internal events = 1.1E-7/yr (Section 19.1.4.2.2.1). The LRF from at-power fire = 1.7E-7/yr (Section 19.1.5.2.2). The LRF from at-power internal floods = 1.7E-8/yr (Section 19.1.5.3.2). The LRF from LPSD internal events = 1.2E-7/yr (Section 19.1.6.2.2.1). The LRF from LPSD fires = 1.3E-7/yr (Section 19.1.6.3.2). The LRF from LPSD internal floods = 1.8E-8/yr (Section 19.1.6.4.2.1). Therefore, the total LRF from all contributors = 5.65E-7/yr.	Jeff Leary
PRA-26	Verification / justification	Page 19.1-198 states "These probabilities [for failure of hydrogen control] are believed to be conservative, but additional calculations are needed to confirm." What is the basis for stating these probabilities are conservative? When will the additional calculations be completed?	Jason Schaperow	The hydrogen calculations to support the assumptions were not complete at the time of the initial submittal, so this key assumption was made to bound the effect of hydrogen-induced containment failure. The calculations are now complete and there results will be incorporated into the analysis. The results confirm that the assumptions made were conservative, and the updated analysis will either retain the conservative assumptions or refine the numbers to be more realistic. In either case, the revised analysis will present the results of the detailed hydrogen calculations.	Jeff Leary
PRA-27	Verification / justification	DCD Table 19.1-14 shows the basic event failure rate for I-ATWS-RPMCF as 2.98E-07 per day whereas NUREG/CR-6928 reports this failure rate as 2.98E-07 per hour. Does it affect the PRA?	Tony Nakanishi	Event I-ATWS-RPMCF is used for failure to scram due to mechanical reasons. After the onset of the transient, if the control rods fail to insert upon a demand, the event immediately becomes an ATWS event. Although the rods need to insert immediately, the mission time for I-ATWS-RPMCF was assumed to be 1 hr. Note that the original version of NUREG/CR-6928 (published February 2007) considered this event to be a demand failure.	Greg Rozga
PRA-28	Verification / justification	Section 19.1.4.1.1.1, it is unclear how the initiating event frequencies are quantified and where it is documented.	Ayo Ayegbusi	Section 19.1.4.1.1.1 give a relatively generic description of how initiating events are calculated. Details of the calculation of the IE frequencies are part of the initiating event analysis. The initiating events frequencies are developed by either quantifying fault trees (support system initiating events), generic data (NUREGs), modified generic data (SLOCA), and additional PRA analysis calculations (ISLOCA). See Initiating Event Analysis Notebook (APR1400-K-P-NR-013101-P) for further details.	Steven Phillippi
PRA-29	Verification / justification	Section 19.1.4.1.1.1, it is unclear whether any other references are used to calculate IE frequencies apart from NUREG/CR-6928.	Ayo Ayegbusi	An additional generic data reference used (NUREG-1829) is identified in Table 19.1-6 as the value for initiating event RVR. This is an individual exception to the other initiating events that use generic plant data for the initiating event frequency or are identified in Table 19.1-6 as being calculated by other means. Other IE frequencies are calculated by quantifying fault trees or performing additional PRA analysis calculations. See Initiating Event Analysis Notebook (APR1400-K-P-NR-013101-P) for further details.	Steven Phillippi

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PRA-30	Verification / justification	Section 19.1.4.1.1.2, the staff could find no evaluation of plant response and how it was performed.	Ayo Ayegbusi	Section 19.1.4.1.1.2 contains a discussion of the development of the accident sequence analysis. The resultant event trees (which are time sequenced accident progressions) are shown in Figures 19.1-15 through 19.1-39. Table 19.1-8 contains a description of the event tree top events and associated success criteria. Furthermore, Tables 19.1-12 and 19.1-13 summarize the plant response presented in the RELAP and MAAP thermal hydraulic run analysis. A detailed description of each accident sequence progression, associated success criteria, and thermal hydraulic analysis is not provided in the DCD. This detailed information is provided in the PRA model documentation (primarily in the Success Criteria and the Accident Sequence notebooks); the PRA model documentation is referenced by the PRA Summary Document (which is Reference 7 of the DCD as stated in Section 19.1 of the DCD).	Joe Lavelline
PRA-31	Verification / justification	Section 19.1.4.1.1.2, in the RCS heat removal section, the word "may" is used for the feed and bleed operation. It is unclear whether which situations are considered and what they are.	Ayo Ayegbusi	The sentence under Item e. should be reworded to state: "The feed and bleed operation is also able to perform this function."	Joe Lavelline
PRA-32	Verification / justification	Section 19.1.4.1.1.3, it is unclear what is meant by "based on the ASME standard."	Ayo Ayegbusi	The first sentence under this section should be reworded to state: "The approach used in this success criteria analysis is designed to address the supporting requirements of the ASME/ANS PRA Standard for the success criteria technical element."	Joe Lavelline
PRA-33	Verification / justification	Section 19.1.4.1.1.3, it is unclear which success criteria were performed using RELAP and which ones were not.	Ayo Ayegbusi	The at-power Success Criteria notebook identifies the use of RELAP for 12 Large LOCA scenarios, 8 Medium LOCA, 13 Small LOCA, and 4 SGTR scenarios. The MAAP code was also used for some of these initiating events and for other initiating events, but RELAP was chosen for the more complex analyses. This is documented in Tables 5-1 and 5-3 in the SC notebook (APR1400-K-P-NR-013103-P) in detail.	Young In
PRA-34	Verification / justification	Section 19.1.4.1.1.3, it is unclear what is included in the emergency operating guidelines (EOGs) and where they are documented.	Ayo Ayegbusi	The emergency operating guidelines (EOGs) are available. The document can be made available in ERR for review, if necessary. The contents of the EOG are as follows; - STANDARD POST TRIP ACTIONS - DIAGNOSTIC ACTIONS - REACTOR TRIP RECOVERY GUIDELINE - OPTIMAL RECOVERY GUIDELINE; . LOSS OF COOLANT ACCIDENT RECOVERY GUIDELINE . STEAM GENERATOR TUBE RUPTURE RECOVERY GUIDELINE . EXCESS STEAM DEMAND EVENT RECOVERY GUIDELINE . LOSS OF ALL FEEDWATER RECOVERY GUIDELINE . LOSS OF OFFSITE POWER RECOVERY GUIDELINE . STATION BLACKOUT RECOVERY GUIDELINE - FUNCTIONAL RECOVERY GUIDELINE	Char Ryerson
PRA-35	Verification / justification	Section 19.1.4.1.1.3, it is unclear what is meant by the representative results. Are the results provided not the results from the analysis performed?	Ayo Ayegbusi	Under Item k, The word "representative" should be removed from this sentence.	Joe Lavelline

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PRA-36	Verification / justification	Section 19.1.4.1.1.7, it is unclear how the dependencies between HFEs are assessed and how 'appropriate' is determined.	Ayo Ayegbusi	HRA notebook (APR1400-K-P-NR-013105-P) provides the dependency analysis detailed discussion. Will remove 'appropriate' word from the DCD.	Char Ryerson
PRA-37	Verification / justification	In Table 19.1-22, it is unclear why item rank #1 has a significantly higher RAW value than item rank #2.	Courtney St. Peters	The importance values are actual values from the analysis, where the reason is the common cause failure of reactor trip breaker of RPS and DPS.	Joe Lavelline
Completeness - Information that may need to be provided or docketed to support a reasonable assurance finding.					
PRA-38	Completeness	Section 19.1.1 "Uses and Applications of the PRA," the uses of PRA in the design process are not specifically described.	Hanh Phan	The process is established, and the procedures are available. There were several cases where the designs were actually changed (for example, IA, EDG, AAC GTG, cable protection from fire, etc.).	Young In
PRA-39	Completeness	Section 19.1.3, "Special Design/Operational Features," the staff could find no tables or descriptions of the dependencies between front line systems and support systems interfacing and also the dependencies between support systems and support systems interfacing.	Courtney St. Peters	A table for the dependencies between front line systems and support systems is available, but not the dependencies between support systems and support systems interfaces. The tables can be provided, if necessary.	Joe Lavelline Kim Kisoo
PRA-40	Completeness	Section 19.1.4.1.1.1, "Initiating Events" and Section 19.1.4.1.1.3, "Success Criteria Analysis," the staff could not find the details of evaluations/analysis performed to support the information provided in the initiating events and success criteria sections.	Ayo Ayegbusi	Section 19.1.4.1.1.1 and 19.1.4.1.1.3 contain a discussion of the development of the initiating event and success criteria analysis, respectively. Tables 19.1-6 address the relationship between the plant safety functions and initiating events. Table 19.1-6 contains the internal events PRA initiating event frequencies. Tables 19.1-10 and 19.1-11 contain dependency matrices related to initiating events. Table 19.1-8 describes the success criteria for event tree tops. A detailed description of the initiating event and success criteria is not provided in the DCD. This detailed information is provided in the PRA model documentation (primarily in the Initiating Events and Success Criteria Notebooks); the PRA model documentation is referenced by the PRA Summary Document (which is Reference 7 of the DCD as stated in Section 19.1 of the DCD). See Initiating Event Analysis Notebook (APR1400-K-P-NR-013101-P) and Success Criteria Notebook (APR1400-K-P-NR-013103-P) for further details.	Joe Lavelline
PRA-41	Completeness	Section 19.1.4.1.1.2, "Accident Sequence Analysis," the staff could find no discussions on event tree development and related assumptions.	Ayo Ayegbusi	Section 19.1.4.1.1.2 contains a discussion of the development of the accident sequence analysis. The resultant event trees (which are time sequenced accident progressions) are shown in Figures 19.1-15 through 19.1-39. Table 19.1-8 contains a description of the event tree top events and associated success criteria. A detailed description of each accident sequence progression is not provided in the DCD. This detailed information is provided in the PRA model documentation (primarily in the Accident Sequence Analysis Notebook); the PRA model documentation is referenced by the PRA Summary Document (which is Reference 7 of the DCD as stated in Section 19.1 of the DCD).	Joe Lavelline

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PRA-42	Completeness	Section 19.1.4.1.1.6, "Human Reliability Analysis," the staff could not find the human failure events and provide a defensible basis to support its probability estimate.	Courtney St. Peters	Section 19.1.4.1.1.6 contains a discussion of the development of the human reliability analysis. A detailed description of the development of each modeled human action and the associated basis is provided in the HRA Notebook; the PRA model documentation is referenced by the PRA Summary Document (which is Reference 7 of the DCD as stated in Section 19.1 of the DCD). Inclusion of tables of pre-initiators and post-initiators (with critical attributes displayed) should be considered.	Joe Lavelline
PRA-43	Completeness	Section 19.1.4.1.2.5, "Key Assumptions," may not include all important assumptions in the PRA development and external hazards assessments.	Hanh Phan	Each PRA notebook contains detailed assumptions relevant to each technical element.	Joe Lavelline
PRA-44	Completeness	Sections 19.1.4, 19.1.5, and 19.1.6, "Safety Insights from Internal Events PRA, External Events PRA, and Low Power and Shutdown PRA," a comprehensive list of risk-insights derived by PRA and other quantitative assessments was not found.	Hanh Phan	Various subsections within Sections 19.1.4, 19.1.5, and 19.1.6 entitled "Risk Insights" contain a listing of risk insights for various hazard groupings. Furthermore Tables 19.1-3 and 19.1-4 address this issue.	Joe Lavelline
PRA-45	Completeness	Section 19.1.5.3, "Internal Flooding Risk Evaluation" and Section 19.1.6.4, "Internal Flooding PRA for Low Power and Shutdown Operations," the staff could not find a list of flooding sources that were evaluated along with a basis for each screened out flooding source, along with the equipment assumed to be affected in each flood area.	Tony Nakanishi	All flood sources considered in the at-power flooding PRA were reviewed for impact to a loss of shutdown cooling and rescreened. These analyses are documented in the LPSD internal flooding notebooks (APR1400-K-P-NR-013759-P).	Ray Dremel
PRA-46	Completeness	Section 19.1.6, appears to be missing event trees for all plant operational states other than plant operational state 5.	Marie Pohida	Notebook APR1400-K-P-NR-013702-P includes 138 event trees, one for each initiating event, in each applicable Plant Operating State, in Appendix A.	Ross Anderson
PRA-47	Completeness	Section 19.1.6, the staff acknowledges that the reactor cavity is filled to the level necessary for core alterations in POS 7 and POS 9. The staff also acknowledges the flow limitations in the letdown line. However, to screen POS 7 and POS 9 from the PRA, the staff could not find: (a) an evaluation documenting the time to core damage given an extended loss of the decay heat removal function and (b) an evaluation that considers all possible drain paths from the refueling cavity including: potential drain rates, the availability of instrumentation and alarms to detect and mitigate a potential drain path, the likelihood of the operator failing to terminate the leak path, and the availability of SSCs to restore RCS inventory.	Marie Pohida	The time to core uncovery or damage during cavity-flooded conditions, following a loss of decay heat removal, was not calculated. The LPSD initiators include the JL event, which is a rupture of the purification (letdown) line. This event has been classified as "unrecoverable" because a rupture cannot be isolated without also isolating the shutdown cooling system. The JL event can be identified by decreasing level in the reactor cavity; sump or radiation alarms in the vicinity of the rupture; or suction faults (flow, pump amps) for the Shutdown Cooling System. Discovery may occur due to instrumentation alarms or operator observation. There will be a full operations shift crew in the plant, monitoring control room instrumentation and the refueling operations. All operators are trained to watch for these abnormal conditions and take corrective action. Each RCS hot leg has separate letdown lines to the Shutdown Cooling System. A rupture may occur at any point along these lines. If unisolated, a rupture in one line can drain inventory until suction from the hot leg is lost. These divisions are not cross-tied, however, so that a rupture on one line may potentially be isolated, with recovery via the opposite division. Failure of the cavity seal can unrecoverably lower level to the vessel flange. No Human Error Probability was calculated for operator isolation of a rupture with the cavity flooded. Given the likely small size of a low pressure/low temperature rupture and the large inventory involved, a failure probability on the order of 1E-4 is expected. Makeup inventory is available via the IRWSTs and may be delivered via an intact loop with either a Safety Injection or a Shutdown Cooling pump. The	Ross Anderson

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-48	Completeness	Section 19.1.6, the staff understands POS 12B was screened based on thermal-hydraulic analysis, assuming the time to core damage is greater than 24 hours after a loss of shutdown cooling. However, losses of inventory occurring in POS 12B may result in core damage occurring before 24 hours. The staff could not find the quantitatively assessment of POS 12B.	Marie Pohida	Thermal-hydraulic analysis for LOCA case of POS12B has been performed. According to the thermal-hydraulic analysis, core uncovery does not occur within the simulation time of 25 hours in the base case of POS 12B. For the LOCA cases, core uncovery does not occur until 23.7 hours, and core damage does not occur within the simulation time of 25 hours. Therefore, these are screened out.	Ross Anderson
PRA-49	Completeness	To evaluate the risk-insights provided in the DCD, it is unclear whether any single failure that would cause a loss of two trains/divisions was identified.	Hanh Phan	In general, the risk insights do reflect any significant failures that cause a loss of any key safety function through the overall CDF/LRF evaluations.	Young In
PRA-50	Completeness	Section 19.1.2.4, “PRA Maintenance and Upgrade” states that “The APR1400 PRA model and supporting documentation are to be maintained so that they continue to reflect the as-designed characteristics of the plant. Consistent with the ASME/ANS PRA Standard, and NRC RG 1.200, a process is in place to perform the following as applicable to the certified design: a) Monitor PRA inputs and collect any new information relevant to the PRA; b) Maintain and upgrade the PRA to be consistent with the design; c) Consider cumulative impacts of pending changes when applying the PRA; d) Consider impacts of changes for previously implemented risk-informed decisions that used the PRA (e.g., RAP); e) Maintain configuration control of the computational methods used to support the PRA; and f) Document the PRA model and processes.” The staff could find no details and timelines for these activities.	Hanh Phan	RM Procedures: DC-DG-03-24, “Risk Management procedure” EP-6.41, “Risk Management Engineering Configuration Control” EP-6.42, “Risk Management Documentation” EP-6.45, “Risk Management Engineering Training and Certification” EP-6.47, “Risk Management Engineering Peer Review, Independent Review and Self Assessment” RAP Procedure DC-DG-03-09, “Implementation of the Reliability Assurance Program (RAP) DC-DG-03-10, “Expert Panel Roles and Responsibilities” DC-DG-03-11, “Risk Significance Determination of RAP SSCs” EP-6.43, “Risk Management Input to RAP”	Young In
PRA-51	Completeness	DCD Page 19.1-149 states, "Several cables have been identified as requiring fire protection features to prevent damage or spurious operation of related components." It is unclear which SSCs are affected by these cables and how the affected SSCs are modeled in the PRA. The staff found the same statement made on DCD Page 19.1-221 for LPSD fire analysis.	Tony Nakanishi	Since the current cable mapping is based on the reference plant, detailed circuit analysis was not performed, and instead all cables associated with an SSC are conservatively assumed to fail it in the worst possible way with respect to core damage and/or large early release risk. In certain fire compartments, this can become significant. In those cases, key cables were identified. For these key cables, it was noted that either detailed circuit analysis needs to be performed to verify that the cable does not result in the key failure, or the cable requires fire protection features to prevent damage or spurious operation of the related components. The protcted cables are documented in Appendix D of APR1400-K-P-NR-013402-P.	Greg Rozga
PRA-52	Completeness	DCD Page 19.1-221 states, "No fires were identified that can fail both divisions of safety equipment without conditional failure of a fire barrier." This risk insight was identified for the LPSD fire analysis but not for at-power fire analysis.	Tony Nakanishi	The insight is applicable to full power fire as well. Note that the insight was meant to include all plant areas except the Main Control Room (MCR) and the reactor containment building as these are obvious areas where equipment from both divisions exist.	Greg Rozga

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PRA-53	Completeness	It is unclear why some reactor trip sequences do not appear to include the consequential LOOP event. Also, it is unclear from the information in the DCD, how the consequential LOOP was modeled for LOCA and other events that may actuate the ESF.	Tony Nakanishi	Consequential LOOP is evaluated for all initiators. Consequential LOOP is modeled as a top event in the event trees for general transients, loss of condenser vacuum, loss of DC bus A or B, loss of feedwater and loss of instrument air (see top event GRID the event trees in Figures 19.1-22 through 19.1-27). In addition consequential LOOP is modeled in the total loss of CCW and ESW events (see top event GRID the event trees in Figures 19.1-36 and 19.1-38, respectively). It is modeled as a transfer to the "GRID-LOOP" event tree for these events as the LOOP event tree is more indicative of the accident sequence progression. For the other events including feedwater line break, secondary side breaks, partial loss of CCW and ESW, SGTR and LOCAs, the accident sequence progression would be lost if there were a transfer to the GRID-LOOP tree. Hence, the conditional loss of offsite power is modeled within the fault trees supporting the event tree top events for these initiators (fault tree gate GNP-GRID). GNP-GRID, which incorporates the probability of consequential LOOP	Greg Rozga
PRA-54	Completeness	The staff could not find a summary of the uncertainty analysis results for the at-power fire analysis.	Tony Nakanishi	This was not performed in the initial analysis. Will provide the analysis during next update, as needed.	Greg Rozga
PRA-55	Completeness	It is unclear how the house load operation (HLO) described in DCD Page 19.1-20 is modeled or assumed in the PRA.	Tony Nakanishi	DCD Page 19.1-20 describes APR1400 design features for AC Power System. The house load operation (HLO) is not modeled in the PRA.	Young In
PRA-56	Completeness	DCD Page 19.1-260 states, "The room heatup calculations to be supplied are expected to show that room cooling is not required." The staff could not find a summary of the results of these calculations.	Tony Nakanishi	See PRA-8.	Young In
PRA-57	Completeness	DCD Page 19.1-46 states, "A thorough treatment of CCFs, intra-system dependencies, and selected intersystem dependencies is provided." The staff could not find a description of the intersystem dependencies modeled.	Tony Nakanishi	This is documented in the DATA Analysis Notebook (APR1400-K-P-NR-013104-P).	Young In
PRA-58	Completeness	DCD Section 19.1.4.1.1.6 describes the HRA. It is unclear how the HFE patterns were determined for the subsequent dependency analysis.	Tony Nakanishi	The HRA Notebook (APR1400-K-P-NR-013105-P) contains the dependency analysis description.	Char Ryerson
PRA-59	Completeness	Page 19.1-224 states “LPSD flooding LRF is not quantified.” It is unclear why LRF was not quantified for LPSD flooding.	Jason Schaperow	The LPSD flooding CDF was very low (1.8E-8/yr). The choice to conservatively bin all LPSD flooding CDF to LRF was made to minimize the work scope. A detailed analysis is expected to show an LPSD internal flood CPLR similar to that see in the LPSD internal events and fire analyses. The conservative application of all LPSD internal flood CDF to LRF had a very small impact on the total LRF because of the low LPSD internal flood CDF.	Jeff Leary
PRA-60	Completeness	Section 19.1.4.1.1, under data analysis, it does not describe the data sources used for initiating events.	Ayo Ayegbusi	This is documented in Attachment 2 in DATA Analysis Notebook (APR1400-K-P-NR-013104-P).	Steven Phillippi

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-61	Completeness	Section 19.1.4.1.1.1, it is unclear why initiating events in Table 19.1-5 lists only 4 initiating event types, when Table 19.1-6 identifies more than 4 initiating events.	Ayo Ayegbusi	The initiating events identified in Table 19.1-6 can be placed into one of the 4 categories identified in Table 19.1-5. This is further discussed in the initiating events analysis. <u>LOCAS:</u> LLOCA, MLOCA, SLOCA, RVR, & ISLOCA <u>Sec. Piping Breaks:</u> SGTR, LSSB-U, LSSB-D, LOFW, FWLB <u>Transients (& Special IEs):</u> Transient events include all internal initiating events not described in the above categories: LOOP events, SBO. Special IEs are: LOCV, LODCA, LODCB, LOIA, TLOCCW, PLOCCW, TLOESW, PLOESW) ATWS: ATWS	Steven Phillippi
PRA-62	Completeness	What qualitative evaluation is performed for each potential initiating events identified to assess applicability to APR1400 as stated in Section 19.1.4.1.1.1?	Ayo Ayegbusi	Potential initiating events for the APR1400-DC PRA are identified based on generic industry lists of initiating events, review of plant-specific systems and design features, and consideration of system interfaces, spatial interactions, and common cause failures. For each of the potential initiating events identified a qualitative evaluation is performed to assess the applicability of the event to the APR1400-DC design. As many aspects of the APR1400 are similar to existing pressurized water reactor (PWR) designs, a review of information from existing plants is also considered. The initiating events analysis provides a qualitative disposition of each potential initiating event.	Steven Phillippi
PRA-63	Completeness	Section 19.1.4.1.1.1, what is meant by CCF potentials?	Ayo Ayegbusi	For example, in the support system initiating event (SSIE) fault trees, documented in APR1400-K-P-NR-013101-P.	Steven Phillippi
PRA-64	Completeness	Section 19.1.4.1.1.1, what new initiators unique to APR1400 were identified?	Ayo Ayegbusi	The paragraph describing the new initiators unique to APR1400 refers to potential initiating events that are evaluated in the Initiating Events analysis. All potential unique initiators identified are screened out or are identified into one of the initiating event categories identified in the response to PRA-61 above and in APR1400-K-P-NR-013101-P.	Steven Phillippi
PRA-65	Completeness	Section 19.1.4.1.1.1, what special initiators were considered and/or modeled?	Ayo Ayegbusi	The special initiators are evaluated in the Initiating Events Analysis (APR1400-K-P-NR-013101-P). As listed in the response to PRA-61, special IEs are: LOCV, LODCA, LODCB, LOIA, TLOCCW, PLOCCW, TLOESW, PLOESW, and some are evaluated by SSIE fault trees.	Steven Phillippi
PRA-66	Completeness	Section 19.1.4.1.1.1, how were potential initiating events screened from consideration if the frequency was low?	Ayo Ayegbusi	From Section 19.1.4.1.1.1, paragraph 6: Potential initiating events can be screened from consideration if the frequency of occurrence of the event is sufficiently low. The ASME/ANS PRA Standard allows screening of initiating events that have a frequency less than 1 × 10-7 per reactor year and do not involve interfacing systems LOCA (ISLOCA), containment bypass, or RV rupture. <u>No initiating events for the APR1400 PRA were screened based on frequency.</u>	Steven Phillippi

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PRA-67	Completeness	Section 19.1.4.1.1.1, what are the preliminary and final IE grouping definitions?	Ayo Ayegbusi	The preliminary IE grouping is performed by reviewing the list of potential initiating events for the APR1400-DC PRA, including those identified from generic industry lists of initiating events. Similar initiating events (i.e., those expected to have a common core damage accident sequence progression) are grouped together and the event frequencies are estimated. Following ASME/ANS PRA Standard IE-B-1 to IE-B-5 for IE grouping, Table 19.1-6 shows the results of grouping the specific initiating events with respect to the impacts noted in Table 19.1-5. The initiating events analysis specifically addresses how preliminary and final IE groups.	Steven Phillippi
PRA-68	Completeness	Section 19.1.4.1.1.1, what are the existing nuclear power plants referenced?	Ayo Ayegbusi	Although the APR1400-DC reactor is a new design, many aspects are similar to existing pressurized water reactor (PWR) designs, specifically the C-E System 80, so a review of information from existing plants is considered. As the initiating events are generally similar to those of existing nuclear power plants, the frequency for most initiating events can be calculated based on NRC generic estimates for current power plants (i.e., NUREG/CR-6928).	Steven Phillippi
PRA-69	Completeness	Section 19.1.4.1.1.2, how is "postulated disturbance" used in this section?	Ayo Ayegbusi	The phrase "postulated disturbance" means "initiating event".	Joe Lavelline
PRA-70	Completeness	Section 19.1.4.1.1.2, multiple instances where the word potential is used to describe system response. Are the systems designed to respond or not?	Ayo Ayegbusi	The word "potential" is used in this section to indicate that there is a probability that the system will respond as described or that it will fail.	Joe Lavelline
PRA-71	Completeness	Section 19.1.4.1.1.2, it is unclear how top events/gates were identified.	Ayo Ayegbusi	A set of key safety functions must be satisfied in order to prevent core damage; these functions are included as top events. For each initiating event, the key safety functions are reviewed to determine there applicability (e.g., reactivity control is not necessary for Large LOCA since the break size causes voiding in the reactor core which interrupts the nuclear chain reaction and the injection of borated water will keep the reactor subcritical). The specific top events used for each initiator to satisfy each key safety function is determined in the success criteria analysis, APR1400-K-P-NR-013103-P. The accident sequence analysis is developed and documented in the Accident Sequence Analysis notebook APR1400-K-P-NR-013102-P.	Greg Rozga
PRA-72	Completeness	Section 19.1.4.1.1.2, it is unclear how the event sequence model structure is developed and where it is documented and how the results are used.	Ayo Ayegbusi	For each initiating event, the progression of potential scenarios leading to either a safe state or to core damage is modeled using an event tree. A set of key safety functions must be satisfied in order to prevent core damage; these functions are included as top events. The order of system and operator functional responses are generally ordered in the event trees in sequential order based on the timing of the accident scenarios as they develop. The accident sequence analysis is developed and documented in the Accident Sequence Analysis notebook APR1400-K-P-NR-013102-P.	Greg Rozga

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PRA-73	Completeness	Section 19.1.4.1.1.2, it is unclear whether any unique trees were developed and how they would impact the results.	Ayo Ayegbusi	For each initiating event, an event tree is developed to delineate the accident progression, and the systems and operator actions necessary for mitigating core damage. There were no unique initiators or unique circumstances identified which required a unique event tree. Details of the accident sequence analysis is developed and documented in the Accident Sequence Analysis Notebook APR1400-K-P-NR-013102-P.	Greg Rozga
PRA-74	Completeness	Section 19.1.4.1.1.2 does not list the referenced PWR PRA in the reference section. Thus, it is unclear which specific PRAs are being considered.	Ayo Ayegbusi	Section 19.1 gives an overview of the PRA. In Section 19.1, the final sentence states: "The PRA is documented in an extensive set of PRA notebooks, which are cross-referenced in the PRA Summary Report (Reference 7)." This is the reference to the PWR PRA for Section 19.1.4.1.1.2 and for all other sections.	Joe Lavelline
PRA-75	Completeness	Section 19.1.4.1.1.2, it is unclear how the timing and progression of each accident sequence is determined and how it is used.	Ayo Ayegbusi	Section 19.1.4.1.1.2 contains a discussion of the development of the accident sequence analysis. The resultant event trees (which are time sequenced accident progressions) are shown in Figures 19.1-15 through 19.1-39. Table 19.1-8 contains a description of the event tree top events and associated success criteria. A detailed description of each accident sequence progression is not provided in the DCD. This detailed information is provided in the PRA model documentation (primarily in the Accident Sequence Analysis Notebook); the PRA model documentation is referenced by the PRA Summary Document (which is Reference 7 of the DCD as stated in Section 19.1 of the DCD).	Joe Lavelline
PRA-76	Completeness	Section 19.1.4.1.1.2, it is unclear how the containment cooling function modelled in the event trees is used to prevent core damage.	Ayo Ayegbusi	Section 19.1.4.1.1.2 contains a discussion of the development of the accident sequence analysis. The resultant event trees (which are time sequenced accident progressions) are shown in Figures 19.1-15 through 19.1-39; these event trees contain accident sequence nodes that are associated with containment cooling. Table 19.1-8 contains a description of the event tree top events and associated success criteria. A detailed description of each accident sequence progression is not provided in the DCD. This detailed information is provided in the PRA model documentation (primarily in the Accident Sequence Analysis Notebook); the PRA model documentation is referenced by the PRA Summary Document (which is Reference 7 of the DCD as stated in Section 19.1 of the DCD).	Joe Lavelline
PRA-77	Completeness	Section 19.1.4.1.1.2, it is unclear whether there are any cases where the top events are reordered to simplify the event tree.	Ayo Ayegbusi	The order of system and operator functional responses are generally ordered in the event trees in sequential order based on the timing of the accident scenarios as they develop. In selected cases, events may be ordered differently to simplify the event tree structure while retaining the proper functional relationships. The accident sequence analysis is developed and documented in the Accident Sequence Analysis Notebook, APR1400-K-P-NR-013102-P.	Greg Rozga
PRA-78	Completeness	Section 19.1.4.1.1.3, it is unclear whether there are any scenarios where containment failure is considered to cause core damage.	Ayo Ayegbusi	The RBCM sequences, under the Containment Isolation node of the PDS diagram (Figure 19.1-41) lead to the CFBVB CET (Figure 19.1-46). These sequences are the ones in which containment failure can lead to core damage.	Jeff Leary

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PRA-79	Completeness	Section 19.1.4.1.1.3, it is unclear how the models incorporate existing PWR plant experience.	Ayo Ayegbusi	The EOG from the reference plants already incorporates the existing PWR operating experience. Specifically, the HRA was developed based upon the general framework in existing PWR operating procedures (since no detailed operating procedures exist in the design phase).	Joe Lavelline
PRA-80	Completeness	Section 19.1.4.1.1.3, describe the certain phenomenology that RELAP5 cannot model and how engineering judgment is applied to determine the appropriate code.	Ayo Ayegbusi	Containment performance and post-core damage phenomenology cannot be modeled with the RELAP Code; therefore MAAP was utilized for this purpose.	Joe Lavelline
PRA-81	Completeness	Section 19.1.4.1.1.3, it is unclear what margin was used to account for uncertainties in the models.	Ayo Ayegbusi	This statement implies that when initiating events are grouped, the documented success criteria used is that which is most conservative (and thus the interpretation may result in more PRA analytical margin for the initiating events in the group of less severity). It also implies that when there were borderline cases which lack adequate margin, the decision was made to apply a more conservative criteria to account for uncertainties.	Joe Lavelline
PRA-82	Completeness	Section 19.1.4.1.1.3, it is unclear why the upstream and downstream of LSSB are not listed in the full power level 1 PRA initiating events list.	Ayo Ayegbusi	The initiating events could be modified to list sub-entries to more correctly represent the actual LSSB initiating events: **1)** Large Secondary Side Break Upstream of MSIV (inside containment) (LSSB-U) **2)** Large Secondary Side Break Downstream of MSIV (outside containment) (LSSB-D).	Steven Phillippi
PRA-83	Completeness	Section 19.1.4.1.1.3, it is not clear why LODC is not listed as LODCA and LODCB.	Ayo Ayegbusi	LODC could be modified to list sub-entries to more correctly represent the actual LODC initiating events: **1)** Loss of Class 1E 125V DC A (LODCA) and **2)** Loss of Class 1E 125V DC B (LODCB).	Steven Phillippi
PRA-84	Completeness	Section 19.1.4.1.1.3, it is not clear if TLODC was considered during the success criteria determination.	Ayo Ayegbusi	TLODC was not considered, but LODC is specifically considered by LODCA and LODCB where a loss of a single train of DC power causes a reactor trip.	Steven Phillippi
PRA-85	Completeness	Section 19.1.4.1.1.3, it is not clear why RVR is not included in the list of initiating events.	Ayo Ayegbusi	Reactor Vessel Rupture (RVR) could be included in the list of initiating event.	Steven Phillippi
PRA-86	Completeness	The staff could find no CCFs of digital I&C in Table 19.1-23.	Courtney St. Peters	See PRA-2.	Young In

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PRA-87	Completeness	Section 19.1.4, the staff could find no mission times for any key design features.	Courtney St. Peters	This is in the Data Analysis Notebook (APR1400-K-P-NR-013104-P). 2.5 Mission times A 24 hour mission time is generally assumed for each evaluated initiating event. This value is used directly in the basic event calculations for "running failures". An exception is for normally standby components in normally standby systems and diesel generators, for which the failure to run data is divided into failures to run within the first hour (for which a one-hour mission time is used), and failures to run after the first hour (for which a 23-hour mission time is used). Another exception is for control rod and its drive mechanism. The maximum period for 90% insertion of control rod by using control rod drive is 4 seconds when reactor trip signal occurs. If 4 seconds is applied as a mission time for control rod and its drive mechanism, the failure probability is very low and realistic. Even though the actions for recovery and others are considered, the mission time will not exceed 1 hour. Therefore, the mission times for control rod and control rod drive mechanism are assumed 1 hour.	Joe Lavelline
PRA-88	Completeness	The staff reviewed the applicant's definitions of Plant Operational States (POSs) defined in Table 19.1-81, LPSD Plant Operating States. The staff could not find detailed information regarding the POS definitions to review the LPSD PRA results, e.g., for each POS: (1) the anticipated decay heat level- how many hours post shutdown, (2) the size and locations of any RCS vents- other than the reactor vessel head, (3) the time to RCS boiling given a loss of the decay heat removal function, and (4) the time to core uncovery.	Marie Pohida	(1) Durations are listed on the attached worksheet, entitled POS Durations, taken from APR1400-K-P-NR-013700-P, Table 6.1. (2) The Reactor Gas Vent valves are opened in Mode 5. An open manway in POS 4B is "large." (3) Notebook APR1400-K-P-NR-013703-P includes specific analyses results such as the information as follows: - Time to reach SCS operating temperature limit or pressure limit - Time to reach SCS operating level limit - Time to the first lift of LTOP relief valve - Time to the core damage (4) The size and locations of any RCS vents, the time to RCS boiling, and the time to core uncovery are not available in these notebooks.	Ross Anderson Kim Jae Gab
Consistency - Information within the submittal appears to be inconsistent and should be addressed.					
PRA-89	Consistency	DCD Chapter 1, Page 1.9-98, Item II.N states that "PRA covers seismic events, internal fire events, and internal flooding events as well as internal events. The COL applicant is to perform site-specific PRA evaluations to address any site-specific hazards." However, as discussed in DCD Chapter 19, APR1400 PRA does not include seismic events.	Hanh Phan	The comment is correct in that PRA-based SMA was performed rather than a seismic PRA. This fact should be reflected in the text referenced by the comment and also in the associated COL item(s).	Joe Lavelline
PRA-90	Consistency	DCD Page 19.1-164 states, "This design confines flood water to one quadrant up to an elevation of 78 ft." However, Table 19.1-4 Item 50 states, "There are no doors or passageways connecting the divisions of safety-related equipment up to elevation 68 ft in the auxiliary building."	Tony Nakanishi	The level to which flood barriers are sealed varies from 68-feet to 78-feet. In addition, barriers between quadrants, while not designed to be sealed for flooding, may be solid between the 68-feet and 78-feet elevations thereby providing an effective flood barrier. The DCD should be clarified to say the elevation to which flood barriers are credited.	Ray Dremel

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-91	Consistency	DCD Page 19.1-62 states, "the RCP seal LOCA probability, given a total loss of seal cooling and the RCP trip, is assumed to be equal to 1 × 10-3 per pump." It is unclear how this relates to the basic event, SEAL-AFSUC, which appears to have a assumed failure probability of 4E-3.	Tony Nakanishi	1E-3 is per pump, and 4E-3 is combined representation of four RCPs. See PRA-10, also.	Young In
PRA-92	Consistency	DCD Table 19.1-40 provides the results of the LRF sensitivity analyses where the "External Reactor Vessel Cooling is Credited" sensitivity case is the same as baseline. However, DCD Page 19.1-113 states, "External reactor vessel cooling is conservatively not credited in the baseline Level 2 analysis, but is evaluated in a sensitivity analysis."	Tony Nakanishi	The ERVCS was not credited in the baseline analysis. The sensitivity analysis showed a very small increase in the frequency of intact containment. The frequency increased from 1.13E-6/yr in the baseline to 1.14E-6/yr in the sensitivity. The percentage of intact containment report still rounded to the same number as the baseline. The reason that turning on the credit for ERVCS did not make a significant impact is that in the modeling for success of ERVCS, the RCS must be at medium or low pressure, water must be injected to the vessel and/or the cavity, and containment heat removal must be successful. However, with no credit to ERVCS, success of these functions in medium and low RCS pressure sequences usually ends in an intact containment, even without ERVCS operation. The ERVCS provides an additional means for water injection and improves the ability to keep the reactor vessel intact, but the total benefit to intact containment (which is reported as the sum of the intact containment and the intact vessel source term categories) is not significant.	Jeff Leary
PRA-93	Consistency	DCD Table 19.1-137 which provides the LPSD LRF frequencies add up to about 6.6E-8, which is not consistent with the total LRF for LPSD of 1.2E-7.	Tony Nakanishi	Section 19.1.6.2.2.3 notes that because POSs 1-4A and 13-15 estimate LRF using the at-power conditional probability of large release (CPLR), no new insights into the LPSD risk would be gained by performing importance analyses or other detailed results evaluations. Therefore, the importance analysis in Table 19.1-137 actually only includes cutsets from POSs 4B-12A. The title of Table 19.1-137 should state all POSs 4B-12A.	Jeff Leary
PRA-94	Consistency	DCD Table 19.1-148 which provides the LPSD fire LRF frequencies add up to about 1.17E-7, which is not consistent with the total LPSD fire LRF of 1.3E-7.	Tony Nakanishi	The values in the Table 19.1-148 are not correct. The correct LRF by POS are: POS 1 = N/A, POS = N/A, POS 3A = 5.59E-9, POS 3B-JL = 6.83E-9, POS 3B-LX = N/A, POS 3B-other = 2.58E-8, POS 4A-JL = 4.17E-9, POS 4A-LX = N/A, POS 4A-other = 2.46E-10, POS 4B = 7.81E-9, POS 5 = 2.38E-8, POS 6 = 1.05E-8, POS 10 = 1.90E-8, POS 11 = 5.43E-10, POS 12A = 5.26E-9, POS 12B = N/A, POS 13 = 1.55E-9, POS 14 = N/A, POS 15 = N/A, all Main Control Room fires = 3.11E-9, all MCA fires not evaluated in detail in L2 = 1.06E-8. The total LRF for these is 1.25E-7/yr, rounded up to 1.3E-7/yr.	Jeff Leary

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-95	Consistency	Section 19.1.4.1.1.1, it is not clear how the ISLOCA and RV Rupture IE's in Table 19.1-6 are calculated.	Ayo Ayegbusi	Section 19.1.4.1.1.1 paragraph 5 addresses the calculation process of initiating event frequency as a whole, not specifically identifying RVR and ISLOCA frequencies as being determined in a manner differing from the other IEs. Table 19.1-6 however addresses these individual initiating events (RVR and ISLOCA) with Notes (5) and (6), providing more detail how the frequency was determined. **Table 19.1-6 Note (5)** lists Reactor Vessel Rupture frequency (2.90E-08/rcy) was taken from NUREG-1829, Volume 1, Table 7.19, for break sizes > 31 inches (Reference 52). This value was treated similarly to other LOCA frequencies, converting to per reactor critical year by multiplying by 1 rcy/0.9 rcry. **Table 19.1-6 Note (6)** states that the ISLOCA initiating event frequency (/rcy) is taken from calculation. No Error Factor (EF) is calculated for this initiating event frequency and thus an EF of 10 is assumed. It would be possible to add a sentence to Section 19.1.4.1.1.1 specifying other means (other PRA analysis calculation, generic data, etc) for determining an initiating events frequency.	Steven Phillippi
PRA-96	Consistency	Section 19.1.4.1.1.7, it is not clear why this section states MAAP was used to evaluate the success criteria, when the success criteria section says MAAP and RELAP were used.	Ayo Ayegbusi	Both RELAP and MAAP were used. Section 19.1.4.1.1.7 states that RELAP5/MOD3 is used to analyze the thermal-hydraulic behavior of the plant. The thermal-hydraulic analysis is used to determine the success criteria. Hence, both RELAP and MAAP were used to determine the success criteria. Table 19.1-12 provides a summary of success criteria evaluated with RELAP. Table 19.1-13 provides a summary of the success criteria evaluated with MAAP. Additional details are contained in the APR1400 Success Criteria Notebook, APR1400-K-P-NR-013103-P.	Greg Rozga
Typo - Potential editorial error in the submittal that should be addressed.					
PRA-97	Typo	As discussed in DCD Chapter 19, the risk associated with seismic events was evaluated using a qualitative PRA-based SMA, rather than seismic PRA as stated on Page 1.9-98.	Hanh Phan	Will replace "seismic PRA" with "PRA-based SMA".	Joe Lavelline
PRA-98	Typo	DCD Chapter 9, Pages 9.5-144 and 9.5-148 state that “The PRA for external fire events is based on the methodology in NUREG/CR-6850, as described in Chapter 19 of DCD.” However, Chapter 19 only addresses internal fires, not external fire events.	Hanh Phan	The external event "internal fires" is based on NUREG/CR-6850 methodology. The external event "external fires" is not evaluated in this DCD. See the Other External Events Analysis Notebook, APR1400-K-P-NR-013801-P, for further details.	Greg Rozga

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-99	Typo	DCD Page 19.1-198 states, "Failure of hydrogen control … is assumed to yield … another conditional probability of containment rupture due to hydrogen burn of 0.1 or 0.01." It is unclear whether this value is 0.1 or 0.01.	Tony Nakanishi	The hydrogen calculations to support the assumptions were not complete at the time of the initial submittal, so this key assumption was made to bound the effect of hydrogen-induced containment failure. The calculations are now complete and there results will be incorporated into the analysis. The results confirm that the assumptions made were conservative, and the updated analysis will either retain the conservative assumptions or refine the numbers to be more realistic. In either case, the revised analysis will present the results of the detailed hydrogen calculations. The conditions of conditional probability of containment rupture due to hydrogen burn are as follows. (1) 0.1 – Dry reactor cavity and PARs successful (2) 0.01 – Wet reactor cavity, no early H2 burn, Containment Sprays operate (lowering steam concentration); (3) 0.0 – Wet reactor cavity and either early H2 burn (less H2 available for late burn) or no Containment Spray operation (steam quenched) (4) 1.0 – Dry reactor cavity and failure of PARS	Jeff Leary
PRA-100	Typo	DCD Page 19.1-693: The description and the corresponding basic event name do not match for the following: RCPVO-C-201 and RCPVO-A-200.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-101	Typo	DCD Page 19.1-716: The description and the corresponding basic event name do not match for the following: RCPVO-A-200.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-102	Typo	DCD Page 19.1-716: The description and the corresponding basic event name do not match for the following: CSMPM2A-PP01A.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-103	Typo	DCD Page 19.1-1157: The description and the corresponding basic event name do not match for the following: PSAVC-S-032.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-104	Typo	DCD Page 19.1-1164: The description and the corresponding basic event name do not match for the following: PSAVC-S-032.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-105	Typo	DCD Page 19.1-1167: The description and the corresponding basic event name do not match for the following: PSAVC-S-032.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-106	Typo	DCD Page 19.1-1169: FV should be RAW in the title for Table 19.1-142.	Tony Nakanishi	The table title will be corrected.	Joe Lavelline
PRA-107	Typo	DCD Page 19.1-1342: The description and the corresponding basic event name do not match for the following: PSAVC-S-032.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-108	Typo	DCD Page 19.1-1349: The description and the corresponding basic event name do not match for the following: PSAVC-S-032.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-109	Typo	DCD Page 19.1-1352: The description and the corresponding basic event name do not match for the following: PSAVC-S-032.	Tony Nakanishi	The event description will be corrected.	Joe Lavelline
PRA-110	Typo	One label on Figure 19.1-50 is SRF. The abbreviation SRF did not seem to be defined in Chapter 19.	Jason Schaperow	SRF represents the Small Release Frequency.	Jeff Leary

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
PRA-111	Typo	Section 19.1.4.1.1.1, describes the potential initiating events. It is unclear why there would be "potential" initiating events.	Ayo Ayegbusi	In this section "potential" initiating events should be changed to "candidate" initiating events.	Joe Lavelline
PRA-112	Typo	Section 19.1.4.1.1.1, error in Note 1 of Table 19.1-6, it used 'which a' instead of 'which are.'	Ayo Ayegbusi	Change the wording per the comment from "which a" to "which are"	Joe Lavelline
PRA-113	Typo	Section 19.1.4.1.1.6, “Human Reliability Analysis,” Page 19.1-54, Item b states "Human-induced initiating eventsand are not considered in detail." "Eventsand" should be "events."	Courtney St. Peters	Insert a space between "events" and "and" per the comment.	Joe Lavelline
Section 19.2 "Severe Accident"					
SA-1	Technical issue	Section 19.2.3.3.7, “Equipment Survivability,” the pressure and humidity for equipment survivability are not provided and there are cases where equipment required for severe accident mitigation is not shown to survive.	Jason Schaperow	For equipment survivability, 100% of relative humidity is considered by engineering judgment. The pressure for equipment survivability is 109 psig (123.7 psia), which is considered Factored Load Category, based on the AICC pressure. The detailed information for AICC pressure is incorporated in Severe Accident Analysis Report (APR1400-E-P-NR-14003-P), main body subsection 3.1.3 and Appendix A, subsection 4.1.	KEPCO E&C SA
SA-2	Technical issue	10 CFR 52.47(a)(23) requires for light-water reactor designs to provide a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by phenomena including hydrogen combustion. Section 19.2.3.3.2 Hydrogen Generation and Control of APR1400 DCD Rev. 0 describes hydrogen igniters but the sources of power to the igniters is not mentioned. The staff could not find the sources of power to the igniters in the DCD and the redundancy of power available to the igniters.	Hanry Wagage	The igniters are powered by Class 1E Bus, EDG, AAC DG and Non-class 1E dedicated DC battery. Please refer to the DCD subsection 6.2.5.2.1.	KEPCO E&C SA
SA-3	Technical issue	10 CFR 52.47(a)(23) requires for light-water reactor designs to provide a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by phenomena including high-pressure core melt ejection. Section 19.2.3.3.4.1.2 of APR1400 DCD Rev. 0 states that “the fraction of the dispersed corium that enters the upper containment via the RPV annulus is given by the ratio of the area of RPV annulus, 1.96 m2, to the total flow area, which is the sum of the area of PRV and the area of reactor cavity, 23.76 m2, or 0.082.” No description or analysis is provided.	Hanry Wagage	The detailed description and analysis are provided in Severe Accident Analysis Report (APR1400-E-P-NR-14003-P), Appendix C-1, subsection 4.1.4. For the APR1400 containment, the mixture of steam, gas, and corium particles flow through available flow paths between the reactor cavity and the upper compartment, and the only flow path that leads directly to the upper compartment without significant de-entrainment is the reactor pressure vessel (RPV) annulus, as described in section 19.2.3.3.4.1.2. Thus, the fraction of the dispersed corium that enters the containment dome can be calculated, as follows; $f_{dom} = A_{an} / (A_{an} + A_{cav})$ where, A_{an}: cross-sectional flow area for the RPV annulus A_{cav}: horizontal cross-sectional flow area for the reactor cavity. For the APR1400 containment, the cross sectional flow area for the RPV annulus and the reactor cavity, and the fraction of the dispersed corium that enters the containment dome were A_{an} = 1.96 m^2, A_{cav} = 21.8 m^2, and f_{dom} = 0.082, respectively.	KEPCO E&C SA

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Item #	**Category**		**NRC Reviewer**	**KHNP Response**	**KHNP Engineer**
SA-4	Verification / justification	Section 19.2.6, "Consideration of Potential Design Improvements Under 10 CFR 50.34(f)" and technical reports, APR1400-E-P-NR-14006-P and APR1400-E-P-NR-14006-NP, "SAMDAs for the APR1400," the staff could find no justification to support the applicant's conclusion that risks from external hazards (i.e., tornadoes, high winds, external flooding, etc.) are negligible. Also, the staff could not find the cost estimates for design improvements that are needed for cases where the applicant determined that the benefit calculated from the improvement would be lower than the cost of the improvement.	Jason Schaperow	1. The risk from other external events was determined to be negligible. Therefore, any SAMA to reduce the risk from other external events was concluded to have a negligible benefit. The DCD can be revised to state these conclusions. 2. Detailed cost estimates were not performed because all items indicated a very small benefit that was clearly less than the cost to implement the associated item. Cost estimates and descriptions of the changes considered in the estimates can be provided.	Ray Dremel
SA-5	Verification / justification	Section 19.2.2.5 of APR1400 DCD Rev. 0 states that "This SCS line design satisfies the ISLOCA acceptance criteria because all sections of the system and interfaces are designed to withstand full RCS operating pressure, or they have leak-test capabilities, valve position indications in the control room that function even when isolation valve operators are de-energized, and high-pressure alarms to warn operators when pressure is approaching the design pressure. Deletion of the interfaces from the SCS lines eliminates the potential for an ISLOCA without adversely affecting the performance or operations of the SCS." It is unclear why the first sentence implies the presence of interfaces between SCS and RCS while the second sentence states deletion of such interfaces.	Hanry Wagage	Section 19.2.2.5 of the DCD Rev. 0 will be revised as follows: "The SCS line design satisfies the ISLOCA acceptance criteria because all sections of the system and interfaces are designed to withstand full RCS operating pressure. Deletion of unnecessary interfaces, such as the purification return line, eliminates the potential for an ISLOCA without adversely affecting the performance or operations of the SCS. These design features satisfy the ISLOCA acceptance criteria for the SCS lines."	KEPCO E&C SD
SA-6	Completeness	Table 19.2.3-1 Hydrogen Control System Design Status of APR1400 DCD Rev. 0 identifies locations of only 2 of 8 igniters and 23 of 30 PARs. The staff could not identify locations of all the igniters and PARs on the table.	Hanry Wagage	The detailed location of Hydrogen Control System including 8 igniters and 30 PARs are provided as shown in Figure 19.2.3-1 in DCD. Following igniters / PARs should be inserted in Table 19.2.3-1. - SG #1 Compartment : 2 igniters, 2 PARs - SG #2 Compartment : 2 igniters, 2 PARs - Pressurizer : 2 igniters, 1 PAR	KEPCO E&C SA
Section 17.4 "RAP"					
RAP-1	Technical issue	Section 17.4, Contrary to Table 1.9-2, DCD Section 17.4 does not follow the guidance in SRP 17.4, Rev. 1.	Ayo Ayegbusi	Need to clarify this comment with the staff.	Young In
RAP-2	Technical issue	Section 17.4, the DCD discussion does not meet the guidance in SRP 17.4.A.2.2, "Design Control," where the applicant should describe design change control processes relating to plant changes (not just the RAP list), quality controls of D-RAP inputs, controls of procedures and instructions and records.	Ayo Ayegbusi	RAP procedures are in place. DC-DG-03-09, "Implementation of the Reliability Assurance Program (RAP) DC-DG-03-10, "Expert Panel Roles and Responsibilities" DC-DG-03-11, "Risk Significance Determination of RAP SSCs" EP-6.43, "Risk Management Input to RAP"	Young In
RAP-3	Technical issue	Section 17.4, the DCD does not discuss a rationale for excluding certain types of risk significant SSCs from the RAP such as pipes and ducts nor does the DCD address how other programs and requirements ensure these SSCs do not degrade as provided in SRP 17.4.A.3.	Ayo Ayegbusi	The DCD will be clarified to include these SSCs specifically. The current analysis implies the inclusions.	Young In
RAP-4	Technical issue	Section 17.4, the RAP list does not identify the dominant failure modes of RAP SSCs as provided in SRP 17.4.A.6.	Ayo Ayegbusi	This will be added in the DCD Table 17.4-1.	Young In

POS	Description	Duration (hrs)	HRs after SD	Dys after SD
1	Low power and subcritical operation	3.6	0	0
2	SG cooldown	32.9	3.6	0.2
3a	SCS cooldown to 212	4.6	36.5	1.5
3b	SCS cooldown below 212	37.6	41.1	1.7
4a	Draindown with manway closed	1.3	78.7	3.3
4b	Draindown with manway open	20.3	80	3.3
5	Midloop operations	16.8	100.3	4.2
6	Refill	54.9	117.1	4.9
7	Offload	72	172	7.2
8	Defueled	96	244	10.2
9	Onload	72	340	14.2
10	Draindown	85.7	412	17.2
11	Reduced inventory with manway closed	13.2	497.7	20.7
12a	Refill with manway open	4.2	510.9	21.3
12b	Refill with manway closed	23.2	515.1	21.5
13	Heatup with SCS	33.5	538.3	22.4
14	Heatup with SGs	42.7	571.8	23.8
15	Startup	42.4	614.5	25.6