Response to

Request for Additional Information No. 2, Supplement 1, Revision 0

4/10/2008
U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19
SPLA Branch

Question 19-01:

Section 19.0.1, “NRC Regulatory Requirements and Related Policies,” does not include several items listed in Section 19.0 of the Standard Review Plan (SRP). For completeness, please confirm in Section 19.0.1 that the following requirements and guidance either were considered or are not applicable: (1) Title 10 of the Code of Federal Regulations (10 CFR) 52.47(a)(8); (2) 10 CFR 52.47(a)(23); (3) 10 CFR 52.47(a)(27); (4) NRC Policy Statement, “Regulation of Advanced Nuclear Plants”; (5) SECY-96-128 and the related Staff Requirements Memorandum (SRM); (6) SECY 97-044 and the related SRM.

Response to Question 19-01:

Applicability of SRP criteria is documented in ANP-10292, “U.S. EPR Conformance with SRP Acceptance Criteria,” AREVA NP Inc., December 2007, which was submitted with the U.S. EPR FSAR. Table 19-01-1 summarizes the disposition of the items of interest. The applicability of SECY 97-044 and the related SRM (item 6) will be changed in ANP-10292 to note the use of a non-safety related containment spray system (called SAHRS) to remove heat generated in the containment during a severe accident as recommended in the guidance.

FSAR Impact:

ANP-10292 page 952 is revised as described in the response and shown on the enclosed markup.

Table 19-01-1—Disposition of Licensing Items

Item #	Item	Applicable	Description
(1)	10 CFR 52.47(a)(8)	Yes	FSAR, Tier 2, Table 1.9-3 addresses the disposition of TMI related items associated with 10 CFR 52.47(a)(8). See ANP-10292 item 19.0-AC-01.
(2)	10 CFR 52.47(a)(23)	Yes	ANP-10292 item 19.0 AC-02 addresses the analysis of design features for the prevention and mitigation of severe accidents in accordance with 10 CFR 52.47(a)(23). See FSAR Sections 19.2.2 “Severe Accident Prevention”, 19.2.3 “Severe Accident Mitigation”, and 6.2.5 “Combustible Gas Control in Containment” for more details.
(3)	10 CFR 52.47(a)(27)	Yes	ANP-10292 item 19.0-AC-03, addresses the requirement for a plant specific PRA. COL Item 19.0-1 (See FSAR, Tier 2, Section 1.8) covers this item.
(4)	NRC Policy Statement, “Regulation of Advanced Nuclear Plants”	Yes	ANP-10292 item 19.0-SAC-04 addresses the NRC Policy Statement, "Regulation of Advanced Nuclear Power Plants” and confirms it is applicable to, and was considered in, the development of Chapter 19.

Item #	Item	Applicable	Description
(5)	SECY-96-128 and the related SRM,	See Description	ANP-10292, item 19.0-SAC-08 addresses SECY-96-128 and the related Staff Requirements Memorandum (SRM). Item VII of SECY 96-128 is related to the use of passive autocatalytic recombiners (PARs) for control of hydrogen during a design basis accident. The issues associated with the use of PARs have been resolved since this SECY was written. The U.S. EPR PARs are described in FSAR Section 6.2.5. The remaining issues in this SECY are applicable to a specific reactor design and not applicable to the U.S. EPR.
(6)	SECY 97-044 and the related SRM	See Description	ANP-10292 item 19.0-SAC-09 addresses SECY-97-044 and the related SRM. SECY 97-044 is applicable to a specific reactor design and addresses the sole reliance on passive heat removal systems in mitigating severe accidents. As recommended in this SECY, the U.S. EPR utilizes a non-safety related severe accident heat removal system (SAHRS) to remove heat generated in the containment during a severe accident. The SAHRS is described in FSAR Chapter 19.2.

Question 19-02:

Footnote 8 in RG 1.206, Section C.I.19, Appendix A, states that: "'PRA [probabilistic risk assessment]-based insights' are those insights identified during the DC [design certification] process that ensure that assumptions made in the PRA will remain valid in the as-to-be-built, as-to-be-operated plant and include assumptions regarding SSC [structure, system, and component] and operator performance and reliability, ITAAC [inspections, tests, analyses, and acceptance criteria], interface requirements, plant features, design and operational programs, and others. The usage of the phrase is intended to be consistent with its use in Table 19.59-29 of the AP600 design control document [DCD]." In the AP600 DCD, each insight receives a disposition such as a reference to another portion of the DCD, an ITAAC, or a combined license (COL) information item. Table 19.1-102, "Summary of Insights from the PRA of the U.S. EPR," does not include a similar disposition for each insight to ensure that the assumptions remain valid in the as-to-be-built, as-to-be-operated plant. For example, the diversity of the station blackout diesel generators is an important assumption that must be retained by future COL holders. Please update Table 19.1-102 to include a disposition for each insight and ensure that the table reflects all important assumptions and insights that must remain valid for future plants.

Response to Question 19-02:

FSAR, Tier 2, Table 19.1-102 is updated to include dispositions for each insight and 20 new insights or assumptions are added (21 through 40).

FSAR Impact:

FSAR, Tier 2, Table 19.1-102 will be revised as described in the response and indicated on the enclosed markup. The text for Insight 21 "Nature of the probability distribution for CDF" in Revision 0 of the FSAR is removed from the insights table and placed in revised Section 19.1.4.1.2.7 (See revised FSAR pages 19.1-58 and 19.1-59).

Question 19-05:

The values presented in the “Risk Metrics” sections throughout Chapter 19 appear to be point estimates. NRC guidance, as well as the American Society of Mechanical Engineers (ASME) PRA standard, specifically requests mean values rather than point estimates. (See Regulatory Guide (RG) 1.174, section 2.2.5.5; RG 1.206, Section C.I.19, Appendix A; SRP 19.0, section III; ASME RA-Sb-2005, Supporting Requirement QU-A2b; and RG 1.200, Table A-1, clarification of requirement QU-A2b.)

Also, the uncertainty cases for which diesel generator uncertainty was eliminated are not appropriate. The footnote to the supporting requirement on uncertainty in the ASME PRA standard refers to a 1981 paper (G. Apostolakis and S. Kaplan, “Pitfalls in Risk Calculations,” Reliability Engineering, Vol. 2, pp 135-145, 1981.) that identifies the importance of handling state-of-knowledge dependencies correctly because of the potential understatement of both the mean and variance of a probability distribution. The worst-case underestimation is a system with four redundant components, where the mean is underestimated by a factor of 300 and the 95th percentile by a factor of 25 for the simple example presented in the paper. The high mean for a probability distribution that includes redundant equipment failures is therefore an important insight, not an artifact that should be handled by removing uncertainty distributions on certain components.

Therefore, the final safety analysis report (FSAR) should be revised to reflect:
(a) Mean values for risk metrics (i.e., core damage frequency (CDF) and large release frequency (LRF)) in all appropriate sections,
(b) Any changes to risk insights or importance measures as a result of using the mean value,
(c) Clarified discussion of the impact of redundant equipment with the same state-of-knowledge-based probability distribution,
(d) Discussion of why the impact is different for the internal events, fire, flooding, shutdown and Level 2 PRA elements.

Additionally, the response to this question should include a list of the correlation classes used in the PRA model and the components that are assigned to each class. If there is any difference between the correlation class grouping and common cause grouping (especially for the emergency and station blackout diesel generators), please identify and justify these differences.

Response to Question 19-05:

Response to Question 19-05a:

The FSAR will be revised to report all mean values in the associated uncertainty subsection of the results sections and refer to them in the risk metrics subsections. Reporting the mean values and associated 95% and 5% values in the uncertainty subsection is more appropriate because these values are dependent on a specific uncertainty run (depends on the selection of a “seed” value for the Monte Carlo simulation and on the RS limitation on the number of the runs used to create the distributions). Therefore, they do not represent constant values to be associated with the U.S. EPR PRA model.

Response to Question 19-05b:

Qualitatively, the risk insights or importance measures are not expected to change significantly as a result of using mean values. Quantitative results to this question have not been developed due to the limitations of the PRA software (Risk Spectrum®). Nevertheless, the presented importance measures and contributions still realistically reflect the insights from the U.S. EPR PRA model.

Response to Question 19-05c:

The importance of the redundant equipment and the state-of-knowledge dependencies is limited for the equipment where common cause failures dominate the results. The impact of the redundant equipment is more important in those cases where equipment independent failures are also significant contributors to the results, as in the case of the diesel generators. In the previous evaluation, when EDGs and SBO diesel generators probability distributions were considered completely correlated, the impact of the state-of-knowledge dependencies was very significant. In the results presented in this FSAR Chapter 19 update, based on the different common cause factors (different vendors, locations, cooling and starting systems, fuel supplies) the EDG and SBO diesel generators probability distributions were not consider to be correlated. The change in this assumption has led to a significant reduction in the mean values. The state-of-knowledge dependencies are also discussed in the uncertainty subsections of the FSAR.

Response to Question 19-05d:

The impact of the state-of-knowledge dependencies was less important on the floods and fires, because of a lesser importance of the diesel generators (only consequential LOOP events were considered).

Other than for the diesel generators (as discussed above), the correlation classes used in the U.S. EPR PRA were based on the applicable failure parameters and not on the common cause groups, which is a conservative grouping. Therefore, all pumps, MOVs, check valves, SOVs, pneumatic valves, and circuit breakers with the same failure parameter were considered in the correlated group. The ten largest groups are listed below:

1. Circuit breakers spurious operation: 320 components
2. MOV spurious operation: 280 components
3. SOV spurious operation: 160 components
4. Bus failures: 140 components
5. Analog signal modifier modules: 140 non-self-monitored (NSM) components and 140 (self-monitored) SM components
6. Analog signal multiplier modules: 140 NSM components and 140 SM components
7. Check valves failure to remain open: 140 components
8. Priority modules: 110 NSM components and 110 SM components
9. MOV internal rupture: 110 components

10. Pneumatic valve spurious operation: 100 components

FSAR Impact:

Impacted FSAR pages, with changes incorporated are provided for the following sections (for Internal Events, Floods, Fires, Shutdown, Total –CDF & LRF):

- Risk Metrics
- Uncertainty Analysis
- Uncertainty Figures

The updates are summarized below:

	Risk Metrics Pages	Uncertainty Analysis Pages	Uncertainty Figures
Internal Events - CDF	19.1-50	19.1-58 thru 19.1-61	19.1-5
Floods - CDF	19.1-118	19.1-124, 19.1-125	19.1-13
Fires - CDF	19.1-137	19.1-143, 19.1-144	19.1-18
Shutdown - CDF	19.1-165	19.1-170, 19.1-171	19.1-23
Total – CDF	19.1-175, 19.1-176	19.1-177, 19.1-178	19.1-29
Internal Events - LRF	19.1-98	19.1-105	19.1-9
Floods - LRF	19.1-125	19.1-132, 19.1-133	19.1-14
Fires - LRF	19.1-144	19.1-152	19.1-19
Total – LRF	19.1-176	19.1-178	19.1-30

Shutdown information is provided for CDF only.

Question 19-07:

Please provide system information (including a description and system drawing or fault tree) as assumed in the PRA for the closed cooling water system, auxiliary cooling water system, and operational chilled water system. These systems appear to be modeled in the PRA, but no information on the systems could be found in the rest of the FSAR.

Response to Question 19-07:

The Operational Chilled Water System (OCWS), Closed Cooling Water System (CLCWS), and the Auxiliary Cooling Water System (ACWS) are modeled in the PRA based on representative system designs. FSAR Section 19.1.2.4 describes the U.S. EPR PRA model as an evolving model subject to the applicants PRA maintenance and upgrade program. COL item 19.1-9 listed in FSAR Table 1.8-2 is provided to confirm that significant PRA assumptions on system modeling remain valid. The OCWS, CLCWS and ACWS systems as modeled in the PRA are described below.

1 OPERATIONAL CHILLED WATER SYSTEM

The operational chilled water system (OCWS) is a non-safety system that has the capability to provide chilled water to various plant systems during normal plant operation. The OCWS consists of four trains. Each train's capability is 33% under the most limiting conditions (highest outside temperature).

The major components for each train are:

- Pump (QNA 21/22/23/24 AP 23/33/43/53).
- Chiller Unit (QNA 21/22/23/24 AN 001).

Each of the four chiller units is cooled by the component cooling water system (CCWS) common header (CH). Two units are cooled by CCW CH1, two units are cooled by CCW CH2.

The OCWS is credited in the PRA to provide chilled water to the electrical division of the Safeguard Building ventilation system (SAC). The OCWS is used as a backup when the safety chilled water system (SCWS /QKA) is unavailable for maintenance or fails during operation.

The PRA model assumes that two OCWS trains are sufficient to provide adequate cooling to the electrical rooms during the mission time of 24 hours. The preferred alignment of the OCWS is to supply cooling to electrical Division 2 and 3 rather than Division 1 and 4.

A simplified drawing of the OCWS as modeled in the PRA is provided in Figure 19-07-1.

2 CLOSED COOLING WATER SYSTEM

The closed cooling water system (CLCWS) is a non-safety balance of plant system whose main function is to remove the heat generated by components of the conventional part of the plant via the closed cooling water heat exchangers to the auxiliary cooling water system (ACWS).

The CLCWS system is comprised of three 50% capacity pumps (PGC11/12/13 AP001) connected to a common header which supplies three 50% capacity heat exchangers (PGC16/17/18 AC001).

The CLCWS is credited in the PRA to provide cooling to the main feedwater system (MFWS) and/or startup and shutdown system pumps. Failure of the CLCWS results in a loss of MFW and SSS (LBOP).

A simplified drawing of the CLCWS as modeled in the PRA is provided in Figure 19-07-2.

3 AUXILIARY COOLING WATER SYSTEM

The auxiliary cooling water system (ACWS) is a balance of plant system whose main function is to transfer heat from the closed cooling water heat exchangers to the normal heat sink (NHS). The circulating water system, which is not modeled explicitly in the PRA, provides water from the NHS to the ACWS. Water is used to cool the CLCWS heat exchangers, and then is returned to the NHS.

The ACWS is comprised of two 100% pumps (PCC 11/12 AP001) that provide flow to the CLCWS heat exchangers.

The PRA models the ACWS as a support system for the CLCWS. The failure of NHS is modeled as an undeveloped event that envelops failures of the circulating water system.

A simplified drawing of the ACWS is provided in Figure 19-07-3. Note: In Figure 19-07-3, the ACWS is referred to as conventional service water system and the NHS is referred to as the UHS.

FSAR Impact:

The FSAR will not be changed as a result of this question.

Figure 19-07-1—Operational Chilled Water System Representative Simplified Drawing

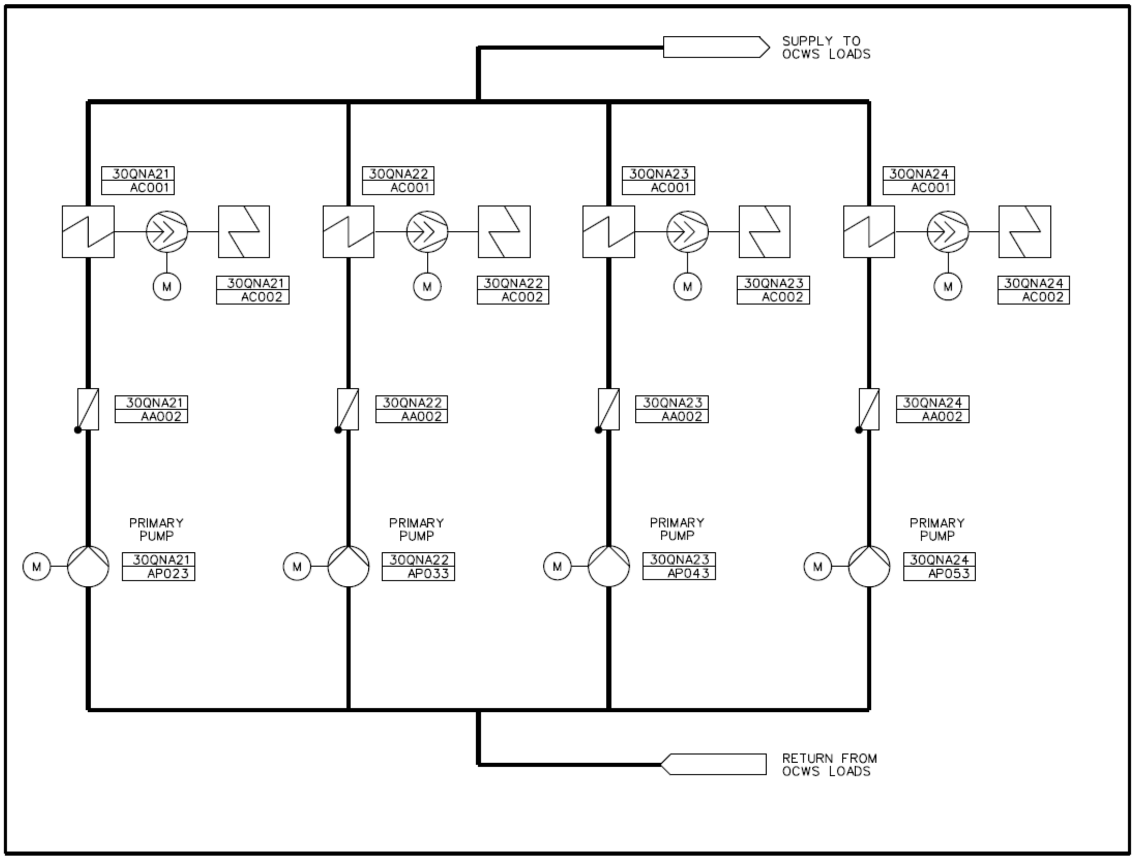


Figure 19-07-2—Closed Cooling Water System Representative Simplified Drawing

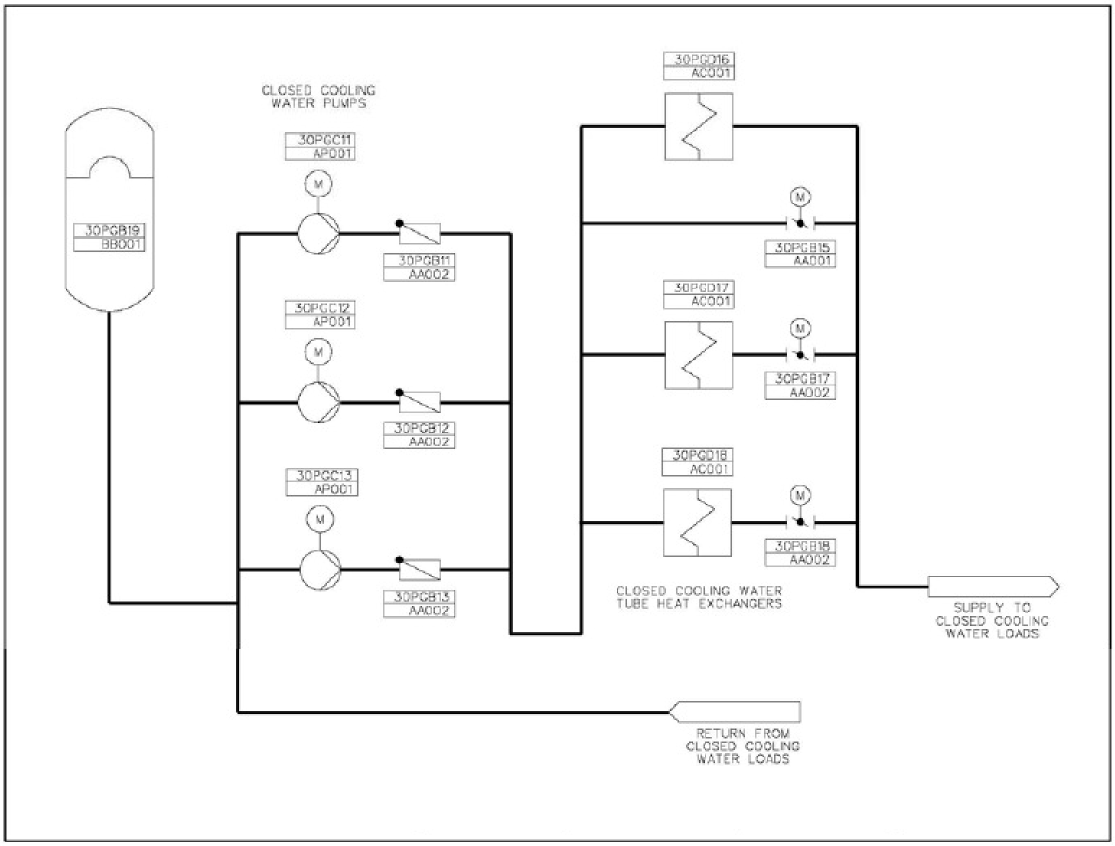
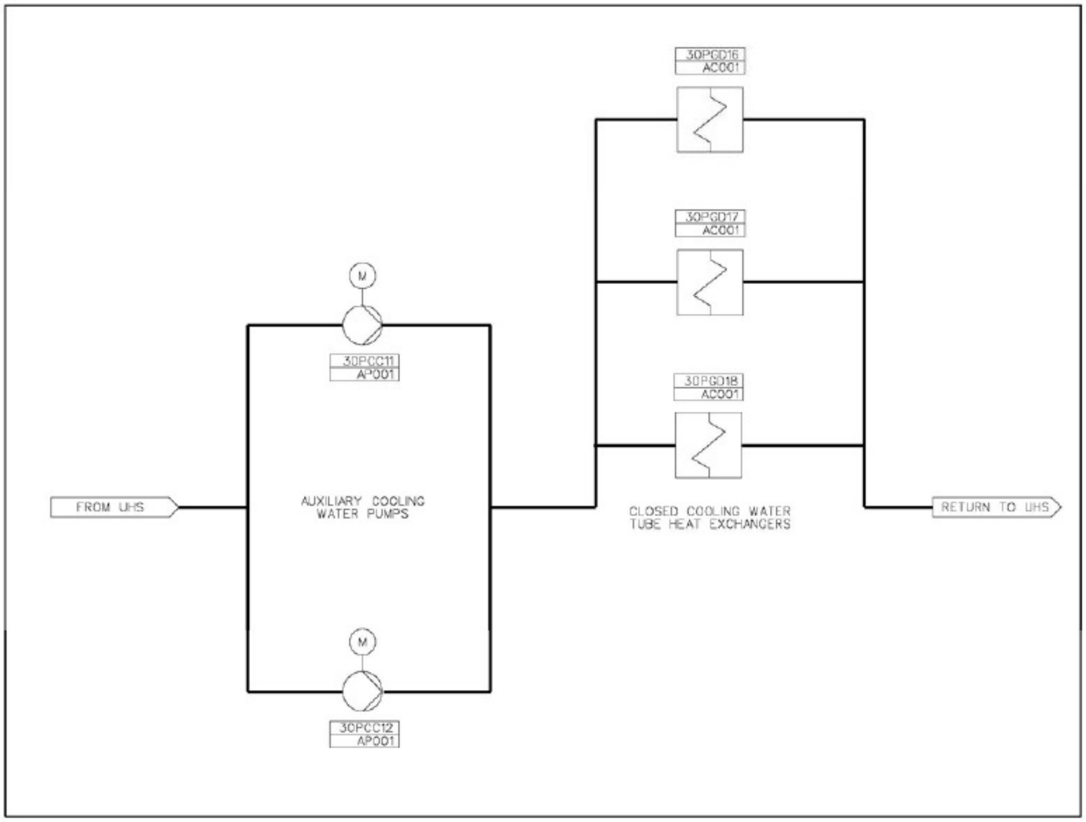


Figure 19-07-3—Auxiliary Cooling Water System Representative Simplified Drawing



Question 19-27:

Sections 5.4.7.2.1 and 19.1.6.1.7 describe design features to address shutdown and mid-loop operations. However, most of these features appear to have limited or no coverage in TS, as presented in the attached table (Table RAI-19-1). Considering this table, please: (a) Confirm the apparent treatment in TS and justify the inclusion or exclusion of the referenced systems and signals according to the four criteria in 10 CFR 50.36. If the analysis determines that the system or signal should not be included in TS, discuss how the availability of these features designed to reduce shutdown risk will be ensured. (b) Discuss how each feature is credited in the shutdown PRA. (c) Provide a sensitivity study for the shutdown PRA that credits only the mitigating systems that are required to be operable according to TS. This request is related to SECY-97-168, in which the staff concluded that the current level of shutdown safety was achieved by voluntary measures that are not required by current regulations, and that these measures could be withdrawn by licensees without NRC approval.

Table RAI-19-1. Treatment of Shutdown Design Features in the U.S. EPR

Design Feature Identified in Section 5.4.7.2.1	Apparent Treatment in Technical Specifications
Inherent redundancy in the design of the four trains safety-related U.S. EPR safety injection system (SIS)/RHRS, with each train having separate RCS connections.	• LCO 3.4.6, 3.4.7, and 3.4.8 address operability of two or three trains of RHR in MODES 4 and 5 • MHSI, in-containment refueling water storage tank (IRWST), CCW, ESW, safeguard building controlled area ventilation system (SBVS), and safeguard building ventilation system electrical division (SBVSED) are only required to be operable in MODES 1-4 per LCO 3.5.3, 3.5.4, 3.7.7, 3.7.8, 3.7.12, and 3.7.13.
Automatic stop of the LHSI [low head safety injection] pumps in RHR mode in the event of a low loop level or low delta-Psat (difference between the RCS hot leg temperature and the RCS hot leg saturation temperature).	• No reference to either signal or to an RHR pump trip caused by either signal could be found in Table 3.3.1-1 of LCO 3.3.1.
Manual opening and closure of the RHR suction isolation valves (in addition of interlocks) prevent unwanted RHR connection or isolation on irregular RCS pressure.	• LCO 3.3.1 requires the inputs to the P14 permissive (wide range hot leg temperature and pressure) in MODE 3, which includes the pressure-temperature condition at which P14 is satisfied.
Safety injection via MHSI with reduced discharge head during low loop level ensures availability of the LHSI pumps for RHR function.	• LCO 3.3.1, Table 3.3.1-1, requires SI manual actuation only in MODES 1-4, does not require an engineered safety features actuation system (ESFAS) signal based on low loop level in any MODE, and does not require a sensor for RCS loop level. • LCO 3.4.11 requires that miniflow lines be open for any MHSI pump capable of injecting into the RCS in MODE 4 (when pressure is less than the low temperature overpressure (LTOP) arming temperature), MODE 5, and MODE 6 (when the reactor vessel head is on).
The RHR connection will be automatically isolated in the event of a break outside of the containment, based on the safeguard building sump level and pressure sensors.	• Section 9.3.3 on equipment and floor drains mentions double sump level measurement in the safeguard buildings, but no indication of a sump level signal or automatic RHR isolation could be found in TS or elsewhere in the FSAR.
Spring-loaded safety relief valve, located at the RHR hot leg suction line, protects the SIS/RHRS against over-pressurization when in RHR mode.	• LCO 3.4.11 does not require RHR suction relief valves as an alternative to pressurizer safety relief valves or an RCS vent.

Design Feature Identified in Section 5.4.7.2.1	Apparent Treatment in Technical Specifications
Redundant hot leg level sensors that initiate RCS make-up when the RCS hot leg has reached low level.	• LCO 3.3.1, Table 3.3.1-1, does not require a sensor for RCS loop level.
During mid-loop operation, the RCS loop level is controlled by the CVCS low pressure reducing valve to ensure there is sufficient RCS water inventory for operation of the LHSI pumps in RHR mode.	• Section 7.7.2.3.13 describes RCS loop level limitation, which is classified as a "control system not important to safety" and is not referred to in either TS or Tier 1.
The reactor pressure vessel (RPV) water level is continually monitored during outage with a level sensor.	• LCO 3.3.1, Table 3.3.1-1, does not require a sensor for RPV water level. • Section 7.1.1.5.7 indicates that the RPV level measurement system uses three temperature sensors at different heights in the hot leg. However, LCO 3.3.1, Table 3.3.1-1, requires hot leg temperature sensors only in MODES 1-3. Cold leg temperature sensors are required in MODES 1-6.
Temperature sensors, located at the RCS hot legs, allow temperature measurement of each hot leg when in a reduced inventory condition.	• LCO 3.3.1, Table 3.3.1-1, requires hot leg temperature sensors only in MODES 1-3. Cold leg temperature sensors are required in MODES 1-6.

Response to Question 19-27:

As stated in the introduction to FSAR Chapter 16, Technical Specifications (TS), the criteria of 10 CFR 50.36 (d)(2)(ii) have been used to identify the structures, systems, components, and design features for which Limiting Conditions for Operation (LCOs) have been included in the US EPR TS. In addition to the requirements established in the TS, the U.S. EPR has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an industry initiative to manage shutdown risk. NUMARC 91-06 provides guidelines for planning and scheduling outage activities in a manner that optimizes required system availability.

Shutdown design features assumed in the Shutdown PRA include all such features and are not limited to only those features governed by TS. Furthermore, shutdown risk will be managed through a combination of technical specifications and administrative controls that will be developed using NUMARC 91-06 as guidance. The NUMARC guidance achieves risk minimization during shutdown not only by ensuring availability of design features (i.e., SSCs) but by limiting and strictly controlling higher risk plant evolutions.

In addition to the above, the TS also enhance the required system availability for plant operation in shutdown and refueling modes. For example, while the TS require the CCW and ESW systems to be Operable in Modes 1 – 4, the TS Bases note that: in Modes 5 and 6, the operability requirements of the (CCW/ESW) System are determined by the systems it supports. Therefore, if an RHR train is required operable in Mode 5, then its respective CCW and ESW trains are also required operable. This constitutes the larger TS definition of operability.

PRA related insights and assumptions are documented in the FSAR as described in the response to question 19-2. COL item 19.1-9 listed in FSAR Table 1.8-2 is provided to confirm that assumptions used in the PRA remain valid.

FSAR Table 19.1-89 provides a high level summary of the systems (See FSAR Table 1.1-1 for a list of system acronyms.) and their associated availability as credited in the shutdown PRA. An expanded version of this table is provided as Table 19-27-1. This table provides additional information on equipment availability and multiple rows for the sub-states that were assumed for POS CA_d. Assumed system availability is based on PRA best estimates of U.S. EPR shutdown progression. The flexibility to remove systems or trains from service and thus manage system availability will be controlled by a combination of TS and administrative controls.

Table 19-27-2 provides a summary of the credited systems and their treatment within the U.S. EPR technical specifications during shutdown conditions. Core damage frequency related risk achievement worth (RAW) is also provided in Table 19-27-2 at a system level. This system level importance provides a measure of the sensitivity to crediting or not crediting the various systems within the U.S. EPR technical specifications.

The RAW results reported in Table 19-27-2 demonstrate that the RAW results for the TS required equipment are dominant. Hence, it would be expected that the results of an integrated sensitivity study would confirm the dominance of the TS required equipment in maintaining an acceptable level of shutdown risk.

In consideration of SECY-97-168, the level of shutdown safety for an advanced LWR such as the U.S. EPR is achieved through a combination of design features, technical specifications and administrative controls. In particular, the U.S. EPR with four independent trains of RHR is unique in this regard.

FSAR Impact:

The FSAR will not be changed as a result of this question.

Table 19-27-1 System Availability During Shutdown

POS	Description	LHSI/RHR Availability				Secondary Cooling Availability					CVCS Pumps		SIS		PSVs & SADVs	CCWS & ESWS	AC & DC Power	SAHR	Hatch
		Trains Avail	RHR Run	RHR Stdby	LHSI Stdby	SG with MSRT	MFW	SSS	MSB & MSIV	EFW	Run	Stdby	Signal Note 4	MHSI					
CA_{d1}	RHR: RCS Normal Level with 2 RHR and SC (shutting down)	4	2 (train 1 & 4)	2 (train 2 & 3)	0 Note 1	4	No Credit	Running	Open	4	2	0	Low delta Psat	4	All	4	4	2	Closed
CA_{d2}	RHR: RCS Solid with 4 RHR and SC (shutting down)	4	4	0	0 Note 1	2	No Credit	No Credit (initially running)	No Credit	2 (Trains 1 and 2 w/ P13)	2	0	Low delta Psat	4	All	4	4	2	Closed
CA_{d3}	RHR: RCS Solid 4 RHR (shutting down)	4	4	0	0 Note 1	2	No Credit	No Credit	No Credit	2 (Trains 1 and 2 w/ P13)	2	0	Low delta Psat	4	All	4	4	2	Closed
CB_d	RHR: Mid-loop w/ RPV head on (shutting down)	4	3	0	1 (train 1 or 4) Note 2	2	No Credit	No Credit	No Credit	2 (Trains 1 and 2 w/ P13)	Train 1	Train2	Low Loop Level	4	All	4	4	2	Closed
D_d	RHR: Mid-loop w/ RPV head off (shutting down)	4	3	0	1 (train 1 or 4) Note 2	NA	NA	NA	NA	NA	Train 1	Train2	Low Loop Level	4	NA	4	4	NA	Closed
E	Cavity Flooded (fuel off load)	3	2 (train 1 & 2)	0 Note 5	1 (train 4) Note 3 and 5	NA	NA	NA	NA	NA	Not Credited	0	Low Loop Level	3 Note 5	NA	3 Note 5	3 Note 5	NA	Open
F	Core Off-load	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	3 Note 5	3 Note 5	NA	NA
E	Cavity Flooded (fuel load)	3	2 (train 1 & 2)	0 Note 5	1 (train 4) Note 3 and 5	NA	NA	NA	NA	NA	Not Credited	0	Low Loop Level	3 Note 5	NA	3 Note 5	3 Note 5	NA	Open
D_u	RHR: Mid-loop w/ RPV head off (after refuel)	4	2 (train 1 & 2)	1 (train 3)	1 (train 4)	NA	NA	NA	NA	NA	1 (train 1)	1 (train 2)	Low Loop Level	4	NA	4	4	NA	Closed
CB_u	RHR: Mid-loop w/ RPV head on (after refuel)	4	2 (train 1 & 2)	1 (train 3)	1 (train 4)	2	No Credit	No Credit	No Credit	2	1 (train 1)	1 (train 2)	Low Loop Level	4	All	4	4	2	Closed
CA_u	RHR: RCS Normal Level (after refuel)	4	2 (train 1 & 2)	1 (train 3)	1 (train 4)	2 to 4	No Credit	No Credit (available or running)	Available	2 to 4	1 (train 1)	1 (train 2)	Low delta Psat	4	All	4	4	2	Closed

1. Model credits operator switchover of Train 1 or 4 from RHR alignment to LHSI
2. Train 1 is running in RHR mode and operator switchover to LHSI is modeled. Also Train 4 is in LHSI alignment; operator switchover to RHR mode is modeled, but no switchover required if LHSI is used. At the end of POS D, LHSI 4 is used to fill cavity halfway and LHSI 3 is used for the last 2 hours to complete cavity fill (during the last 2 hours of LHSI operation, LHSI 4 is assumed to be in maintenance)
3. Trains 1 and 2 are running; Train 4 is aligned to LHSI, but can be realigned to RHR. Train 3 is assumed unavailable (see Note 5)
4. SIS signal changes to Low delta Psat in POS B (P12 permissive). At end of POS CA_{d3}, SIS signal changes to Low Loop Level (P15 permissive)
5. During POS E and F, Train 3 of AC/DC and related systems dependent on this train are assumed unavailable.

Table 19-27-2 System Level Tech Spec Treatment and Importance Measure

System	Tech Spec Section(s)	Tech Spec Treatment	RAW Shutdown
RHR	3.4.6 – Mode 4	Combinations of RCS or RHR loops required Operable	50890
	3.4.7 – Mode 5, Loops Filled	Combinations of RCS or RHR loops required Operable	
	3.4.8 – Mode 5, Loops Not Filled	Two RHR loops operable	
	3.9.4 – Mode 6 – High Water Level	One RHR loop Operable	
	3.9.5 – Mode 6 – Low Water Level	Two RHR loops Operable	
LHSI	See Above	See Above	287
MSRT	3.7.4	Not required operable	<20
MFW	3.7.3	Not required operable	N/A
SSS	Not addressed		N/A
MSIV	3.7.2	Not required operable	N/A
MSB	Not addressed		N/A
CVCS	3.1.8	Letdown isolation Operable in Modes 4, 5, & 6	2099
EFW	3.7.5 3.7.6	One EFW train required in Mode 4; pool volume of 300,000 gal required when SG relied upon in Mode 4	<20
SIS Signal	3.3	Psat not required below Mode 3; Low loop level signal not addressed	8059

Table 19-27-2 System Level Tech Spec Treatment and Importance Measure

System	Tech Spec Section(s)	Tech Spec Treatment	RAW Shutdown
MHSI	3.5.3	Two trains required in Mode 4	752
PSVs	3.4.10	Required for LTOP	<20
SADVs	Not addressed		<20
CCW	3.7.7	Required in Modes 5 and 6 to support RHR operability	558
ESWS	3.7.8	Required in Modes 5 and 6 to support RHR operability	1933
AC Power	3.8.2	One offsite circuit + two EDGs required in Modes 5 & 6	1933
DC Power	3.8.4 3.8.5	Operable in Mode 4 Required in Modes 5 and 6 to support RHR operability	30590
SAHR	Not addressed		N/A
Equipment Hatch	3.9.3	Closed in Mode 4 and in Mode 6 when handling recently irradiated fuel	N/A

Question 19-29:

Table 19.1-102 is missing key U.S. EPR features that reduce shutdown risk and their disposition (e.g., Tier 2, Tier 1, TS, or emergency response guidelines). Please augment this table in the following areas of shutdown risk (the examples are not inclusive): (a) Key design features or structures, systems, and components (SSCs) that reduce the potential of reactor coolant diversion from the vessel through the RHR/CVCS systems (b) Key design features, if any, that automate the response to losses of RHR (c) Key design features, if any, that automate RCS injection following loss of RHR, reactor coolant diversions, and LOCAs (d) Key operator actions and key pieces of instrumentation that are needed to support the associated operator actions (e.g., operator opening a gravity injection flow path) (e) Key SSCs that need to be available at shutdown to provide an alternate decay heat removal path using low pressure makeup and primary pressure relief (f) Key SSCs that are needed to reduce fire risk at shutdown and validate fire risk estimates (e.g., capability of fire watches when fire barriers are not intact)

Response to Question 19-29:

See response to Question 19-2 for an updated list of PRA related assumptions and insights.

Question 19-33:

Please provide a complete electrical dependency matrix including all major accident mitigating systems to supplement Figure 19.1-3. This figure does not include, for example, electrical dependencies of the severe accident heat removal system (SAHRS).

Response to Question 19-33:

Main electrical dependencies for safety systems are as follows: Train 1 is supplied from Electrical Div 1; Train 2 is supplied from Electrical Div 2, etc. Figure 19-33-1 provides a dependency matrix for all safety systems which are not symmetrical (including SAHR), and for non-safety systems as credited in the PRA.

Power assignments for non-safety systems are shown as modeled in the PRA. Changes to the electrical power supplies since the PRA modeling include the following:

- CLCWS Pump 1 power has changed from “Train 1 NPSS AC” to “Train 3 NPSS AC”

- CLCWS Pump 2 power has changed from “Train 2 NPSS AC” to “Train 4 NPSS AC”

FSAR Section 19.1.2.4 describes the U.S. EPR PRA model as an evolving model subject to the applicants PRA maintenance and upgrade program. COL item 19.1-9 listed in FSAR Table 1.8-2 is provided to confirm that significant PRA assumptions on power assignments remain valid.

FSAR Impact:

The FSAR will not be changed as a result of this question.

Figure 19-33-1 —Electrical Dependency Matrix for Selected Systems

	Safety Busses												Non-Safety Busses									
System	Div 1 - EPSS-AC	Div 1 - EUPS-DC	Div 1 - EUPS-AC	Div 2 - EPSS-AC	Div 2 - EUPS-DC	Div 2 - EUPS-AC	Div 3 - EPSS-AC	Div 3 - EUPS-DC	Div 3 - EUPS-AC	Div 4 - EPSS-AC	Div 4 - EUPS-DC	Div 4 - EUPS-AC	Train 1 - NPSS-AC	Train 1 - NUPS-DC	Train 1 – NUPS/12UPS-AC	Train 2 - NPSS-AC	Train 2 - NUPS-DC	Train 2 – NUPS/12UPS-AC	Train 3 - NPSS-AC	Train 4 - NPSS-AC	Train 5 - NPSS-AC	Train 6 - NPSS-AC
EBS – Extra Borating System																						
EBS Pump 1																						
EBS Pump 2																						
Injection to CL1																						
Injection to CL2																						
Injection to CL3																						
Injection to CL4																						
CVCS – Chemical & Volume Control System																						
Charging Pump 1																						
Charging Pump 2																						
Containment Isolation Valves																						
VCT Isolation Valves																						
IRWST Switchover Valves																						
Charging Flow Isolation Valves																						
HP Cooler Isolation Valves																						
SAHR – Severe Accident Heat Removal System																						
SAHR Pump Suction Valves																						
SAHR Pump																						
Pump Discharge to IRWST																						

System	Safety Busses												Non-Safety Busses									
	Div 1 - EPSS-AC	Div 1 - EUPS-DC	Div 1 - EUPS-AC	Div 2 - EPSS-AC	Div 2 - EUPS-DC	Div 2 - EUPS-AC	Div 3 - EPSS-AC	Div 3 - EUPS-DC	Div 3 - EUPS-AC	Div 4 - EPSS-AC	Div 4 - EUPS-DC	Div 4 - EUPS-AC	Train 1 - NPSS-AC	Train 1 - NUPS-DC	Train 1 – NUPS/12UPS-AC	Train 2 - NPSS-AC	Train 2 - NUPS-DC	Train 2 – NUPS/12UPS-AC	Train 3 - NPSS-AC	Train 4 - NPSS-AC	Train 5 - NPSS-AC	Train 6 - NPSS-AC
and Spray Nozzle Isolation Valves																						
SA-CCWS																						
SA-ESWS																						
OCWS – Operational Chilled Water System (QNA)																						
Primary and Secondary Pump																						
Refrigerating Units																						
Flow Control Valves																						
MFWS – Main Feedwater System																						
MFW Pump 1																						
MFW Pump 2																						
MFW Pump 3																						
SSS – Start-up & Shutdown System																						
SSS Pump																						
DWS Makeup (Feedwater Tank Level Control) – Emergency Case																						
MFW/SSS Injection Lines																						
Injection to SG1																						
MOVs																						
SOVs																						
Injection to SG2																						
MOVs																						
SOVs																						

	Safety Busses												Non-Safety Busses											
System	**Div 1 - EPSS-AC**	**Div 1 - EUPS-DC**	**Div 1 - EUPS-AC**	**Div 2 - EPSS-AC**	**Div 2 - EUPS-DC**	**Div 2 - EUPS-AC**	**Div 3 - EPSS-AC**	**Div 3 - EUPS-DC**	**Div 3 - EUPS-AC**	**Div 4 - EPSS-AC**	**Div 4 - EUPS-DC**	**Div 4 - EUPS-AC**	**Train 1 - NPSS-AC**	**Train 1 - NUPS-DC**	**Train 1 – NUPS/12UPS-AC**	**Train 2 - NPSS-AC**	**Train 2 - NUPS-DC**	**Train 2 – NUPS/12UPS-AC**	**Train 3 - NPSS-AC**	**Train 4 - NPSS-AC**	**Train 5 - NPSS-AC**	**Train 6 - NPSS-AC**
Injection to SG3																						
MOVs																						
SOVs																						
Injection to SG4																						
MOVs																						
SOVs																						
ACWS - Auxiliary Cooling Water (PCC)																						
PCC Pump 1																						
PCC Pump 2																						
CLCWS – Closed Cooling Water (PGC)																						
PGC Pump 1																						
PGC Pump 2																						
PGC Pump 3																						

Motive Power
Control Power

Question 19-38:

COL information item 19.1-9 states that the COL applicant must confirm that assumptions used in the PRA remain valid. Several areas for which the modeling is not complete or for which assumptions have been made (such as HVAC recovery times, instrumentation and controls (I&C) details, calibration errors, station blackout human errors, CVCS supply availability, and cooldown operator actions) are in different locations in Chapter 19. Please describe the strategy for communicating these assumptions to the COL applicants (e.g., collection of all assumptions in a single area in the PRA documentation).

Response to Question 19-38:

The communication of assumptions to the COL applicant is in part achieved by the accumulation and presentation of PRA insights and assumptions as described in FSAR Table 19.1-102 (see also response to Question 19-2). It is also achieved by other actions related to COL information items. For example, COL information item 19.1-7 also augments communication of assumptions with respect to seismic margin assessment by requiring the COL to confirm the SMA is bounding for their site.

The transfer of the PRA to the COL Applicant is the point at which the details of the PRA are communicated to COL applicant/holder personnel.

In their COL applications, the COL applicant invokes by reference the DC PRA. The PRA will be updated to keep pace with the successive stages of plant design: detailed design phase, construction phase and operational phase. Information and design details not available at the DC phase will become available with time and be incorporated into the PRA.

AREVA NP is responsible for maintaining and updating the U.S. EPR PRA during the design and construction phases until the PRA ownership/responsibility is assumed by the COL applicant. The time at which this transfer occurs is variable but cannot occur any later than just before the first fuel load (at which time the PRA must be fully updated). The documentation that is supplied at that time is the principal means of communicating all assumptions contained in the PRA.

Prior to that time, PRA assumptions will be communicated both internally and to the COL applicant. The development of emergency operating procedures is an example. PRA personnel will be involved with EOP development to ensure PRA insights and assumptions are considered. The COL applicant will then benefit from the consideration of PRA insights and assumptions when the EOPs are adapted for their specific site.

FSAR Impact:

The FSAR will not be changed as a result of this question.