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U.S. Nuclear Regulatory Commission
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Subject: AmerenUE, NRC Docket No. 52-037
Response to Request for Additional Information for the Callaway Plant Unit 2 RAI No. 1 Revision 0, Section 19.1, Probabilistic Risk Assessment and Severe Accident Evaluation

Reference: Surinder Arora (NRC) to David E. Shafer (AmerenUE), "RAI No. 1 (eRAI No. 1839) - Public" email dated 2/4/09

The purpose of this letter is to respond to the request for additional information (RAI) identified in the NRC e-mail correspondence to AmerenUE, dated 2/4/09 (reference). This RAI addresses the Probabilistic Risk Assessment and Severe Accident Evaluation as discussed in Section 19.1 of the Final Safety Analysis Report (FSAR), as submitted in Part 2 of the Callaway Plant Unit 2 Combined License Application (COLA).

Enclosure 1 provides our response to NRC RAI No. 1 (eRAI No. 1839) - Public, Revision 0. Enclosures 2, 3 and 4 contain proposed COLA changes as a result of the responses contained in Enclosure 1. These responses do not include any new regulatory commitments. COLA impacts associated with each RAI question response are noted in Enclosure 1. This letter provides a partial response to the subject RAI. A supplement to this RAI response will be provided by April 10, 2009 to complete the RAI response associated with question 19-7.

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NRO

If there are any questions regarding this transmittal, please contact Scott Bond at (573) 676-8519, SBond2@ameren.com or Dave Shafer at (573) 676-4722 DShafer@ameren.com.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 2, 2009:

Scott Bond

Scott M. Bond
Manager, New Plants

Enclosure:
1. Response to NRC Request for Additional Information, RAI No. 1 (eRAI No. 1839) - Public, Revision 0; SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation Application Section: 19.1
2. Proposed COLA changes associated with RAI Question 19-1
3. Proposed COLA changes associated with RAI Question 19-2
4. Proposed COLA changes associated with RAI Question 19-6

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Enclosure 1

Response to NRC Request for Additional Information
RAI No. 1 (eRAI No. 1839) - Public, Revision 0;

SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19.1

RAI No. 1 (eRAI No. 1839) - Public, Revision 0;

QUESTION 19-1

The probabilistic risk assessment (PRA) guidance (Chapter 19) in section C.III of Regulatory Guide (RG) 1.206 states that "[i]n cases where it can be shown that assumptions in the certified design PRA (1) bound certain site-specific and plant-specific parameters, and (2) do not have a significant impact on the PRA results and insights, no change to the design certification PRA is necessary." The discussion of losses of offsite power (LOOP) on page 19-8 of the Callaway Plant Unit 2 Final Safety Analysis Report (FSAR) states that "[t]he U.S. EPR PRA Loss of Offsite Power recovery probabilities bound Callaway Plant Unit 2 site-specific values." However, the site-specific values were not provided. Revise the FSAR to include these site-specific values (both at power and during shutdown) and their source.

Response

For a LOOP initiating event, the U.S. EPR one-hour and two-hour LOOP nonrecovery probabilities are provided in Table 19-1-1. These are taken from NUREG/CR-6890 Table ES-3 and are frequency-weighted averages of the four LOOP category industry-average nonrecovery probabilities. Using this same method and substituting the Callaway Plant Unit 1 LOOP category frequencies from NUREG/CR-6890 Table D-1 provides the Callaway Plant Unit 2 LOOP nonrecovery probabilities, also provided in Table 19-1-1.

Recovery from a LOOP that occurs during the 24-hour mission time is incorporated into the basic event LOOP24+REC, provided in Table 19-1-1. The unavailability value for this basic event is calculated by taking the generic at-power LOOP yearly frequency from NUREG/CR-6890 Table ES-1, removing the contribution of consequential LOOPs (consequential LOOPs are handled separately), dividing it by a 24-hour mission time and applying a one-hour nonrecovery probability. Using this same method and substituting the Callaway Plant Unit 1 LOOP category frequencies from NUREG/CR-6890 page D-6 results in the Callaway Plant Unit 2 value provided in Table 19-1-1.

Similarly, the U.S. EPR shutdown LOOP24+REC, provided in Table 19-1-1, is calculated by taking the generic shutdown LOOP yearly frequency from NUREG/CR-6890 Table ES-1, dividing it by 24 hours mission time, and applying a one-hour nonrecovery probability. Shutdown LOOP recovery value is 0.413 and is based on generic data taken from NUREG/CR-6890 Table 4-1. The value is generic and applicable to Callaway Plant Unit 2. This basic event, multiplied by a duration of the specific power operating state, is also used to model LOOP initiators and recoveries in the shutdown states. In conclusion, the use of U.S. EPR™ data for LOOP recovery bounds Callaway Plant Unit 2 site-specific values and the difference does not have a significant impact on the PRA results.

Table 19-1-1: At-Power LOOP Recovery Basic Events

ID	Description	U.S. EPR™ Value	Equivalent Callaway Plant Unit 2 Value
REC OSP 1HR	Failure to Recover Offsite Power Within 1 Hour	5.30E-01	5.16E-01
REC OSP 2HR	Failure to Recover Offsite Power Within 2 Hours	3.18E-01	3.07E-01
LOOP24+REC (at-power)	Loss of Offsite Power During Mission Time and Failure of Recovery Within 1 Hour	4.80E-05	3.95E-05
LOOP24+REC (at-shutdown)	Loss of Offsite Power During Mission Time and Failure of Recovery Within 1 Hour	2.2E-04	2.2E-04

For consequential LOOP (not applicable in shutdown), there are four related basic events, which are provided in Table 19-1-2. The consequential LOOP values (the first column) are taken from NUREG/CR-6890 (page 51), and adjusted for different events. Recovery values (the second column) are taken from the same source (Table A-5). No recovery is credited for a consequential LOOP after a LOCA event. The consequential LOOP values and recoveries are not site-related, they are related to plant events. The values are generic and applicable to Callaway Plant Unit 2.

Note that these values are also provided in the response to U.S. EPR™ RAI Set No. 2, Question 19.01-46.

Table 19-1-2: At-Power Consequential LOOP Recovery Basic Events

ID	Description	Consequential LOOP Value	Non-Recovery Value	Total Value
LOOPCON+REC	Consequential LOOP and Failure of Recovery within 1 Hour for IEs Leading to Auto Scram	5.3E-03	0.33	1.80E-03
LOOPCSD+REC	Consequential LOOP and Failure of Recovery within 1 Hour for IEs Leading to a Controlled Shutdown	5.3E-04	0.33	1.80E-04
LOOPFCSD+REC	Consequential LOOP and Failure of Recovery within 1 Hour for Fire IEs Leading to a Controlled Shutdown	1.1E-03	0.33	3.60E-04
LOOPCONL+REC	Consequential LOOP for LOCA IEs	5.3E-03	1.0	5.30E-03

COL Impact

COLA Part 2, FSAR, will be revised to summarize the response to this question. The changes are shown in Enclosure 2, Proposed COLA changes associated with RAI Question 19-1.

QUESTION 19-2

The discussion of the circulating water system (CWS) on page 19-8 of the FSAR is not detailed enough for the staff to conclude that the U.S. EPR PRA bounds the plant-specific system design. Revise the FSAR to include a quantitative discussion of how the failure probability of the plant-specific CWS and normal heat sink (NHS) is bounded by the NHS undeveloped event modeled in the U.S. EPR PRA, as well as how assumptions related to the NHS model have been confirmed for the Callaway Plant Unit 2 site.

Response

The NHS undeveloped event modeled in the U.S. EPR PRA is "SUP UHS NS". The scope of this undeveloped event, in addition to the NHS, includes the circulating water system ability to provide cooling to the Main Condenser and to supply cooling water to the Auxiliary Cooling Water (ACWS) system. This undeveloped event has an estimated failure frequency of 1.0E-02 per year as part of the Loss of Balance of Plant (LBOP) initiating event and a corresponding failure probability of 2.8E-05 in a 24-hour mission time as a part of the MFW and SSS functional event.

The "SUP UHS NS" undeveloped event failure numbers are based on generic industry data from NUREG/CR-6928 and NUREG/CR-5750. NUREG/CR-6928 provides a Loss of Condenser Heat Sink initiating event frequency of 8.11E-02. NUREG/CR-5750 provides more detailed initiating event data and states that 46% of the Total Loss of Condenser Heat Sink contribution in PWRs is from Loss of Condenser Vacuum. It also states that 36% of the Loss of Condenser Vacuum contribution is from "problems related to the circulating water system: Loss of Non-safety-Related Cooling Water." These values combine to result in a frequency of failure of 1.3E-02 per year. The use of a lesser value of 1.0E-02 per year is considered reasonable because:

- The value of 1.3E-02 per year includes events such as screen plugging, not likely to occur in a closed system. Callaway Plant Unit 2 uses a closed-loop CWS.
- Failures of the CWS and NHS are multiple-counted in the U.S. EPR™ PRA model: They are implicitly included in the generic initiating event frequency for Loss of Main Feedwater (LOMFW) and Loss of Condenser and they are also explicitly included in the ACWS fault tree used in determining the LBOP initiating event frequency.

The values used for the NHS undeveloped event and described above are generic and applicable to Callaway Plant Unit 2.

The Fussell-Vesely importance measure of the "SUP UHS NS" undeveloped event is 1.6E-05: it contributes less than 0.002% to the total CDF.

COL Impact

COLA Part 2, FSAR, will be revised to summarize the response to this question. The changes are shown in Enclosure 3, Proposed COLA changes associated with RAI Question 19-2.

QUESTION 19-3

Discuss whether additional plant-specific changes (other than the ability to use CWS pumps to cool turbine building equipment, as stated in the FSAR) have been made to the PRA models of the closed cooling water system (CLCWS) or auxiliary cooling water system (ACWS), as described in the AREVA NP response to Question 19-07 on the U.S. EPR design certification application.

Response

Regarding the Closed Cooling Water system, the U.S. EPR PRA models this system as 3-50% capacity pumps and 3-50% capacity heat exchangers. This is consistent with the Callaway Plant Unit 2 Closed Cooling Water system design.

The U.S. EPR model of the Auxiliary Cooling Water system differs from the Callaway Plant Unit 2 system. The U.S. EPR PRA models the Auxiliary Cooling Water system as a 1-out-of-2 trains system that takes suction from the Circulating Water system with one pump normally running and one in standby. The Callaway Plant Unit 2 Auxiliary Cooling Water has a similar arrangement with the addition of a bypass around both pumps that allows the Circulating Water system to provide the water supply and motive force for the Auxiliary Cooling Water system. This is the normal mode of operation with both pumps in standby and flow provided by the Circulating Water system. The use of U.S. EPR Auxiliary Cooling Water model bounds Callaway Plant Unit 2 specific system design and the difference does not have a significant impact on the PRA results.

COL Impact

The Callaway Plant Unit 2 COLA will not be changed as a result of this question.

QUESTION 19-4

Discuss how the plant-specific UHS support systems described in section 9.2.5.2 of the Callaway Plant Unit 2 FSAR are modeled in the Callaway Plant Unit 2 PRA. If the support systems are not modeled, demonstrate that the assumptions in the U.S. EPR PRA bound the plant-specific parameters and that there is no significant impact on the PRA results and insights.

Response

Ultimate Heat Sink (UHS) failures are part of the PRA modeling of the Component Cooling Water (CCW) / Essential Service Water (ESW) systems. They are also included in the evaluation of initiating events related to losses of one or two CCW common headers. These CCW common headers are supplied by one of two CCW trains, which are cooled by the ESW trains. The PRA model for the ESW trains does not include failures of the UHS support systems described in the Callaway Plant Unit 2 FSAR Section 9.2.5.2. These support systems are:

- Normal ESW Makeup, supplied from the Raw Water Supply System (RWSS)
- Blowdown from the ESW System Cooling Tower Basins
- ESW Emergency Makeup System
- ESW Makeup Water Chemical Treatment.

The water inventory in two ESW cooling tower basins is sufficient to support the plant operation for 72 hours after a plant trip. Therefore, these support systems have no impact on the ability of the ESW system to perform its function as a mitigating system for the 24-hour mission time following an initiating event.

It is possible that, during plant operation, failures of these support systems could disable one or multiple ESW trains, and lead to an initiating event. However, these events are not likely to have significant impact on the plant risk based on the following insights:

- Even if a failure of the RWSS disabled the normal ESW makeup, there would be extensive time available for the operators to recover the function using the emergency ESW Emergency Makeup System.
- In the unlikely event where the ESW Emergency Makeup System would also fail, a safe shutdown of the plant could still be successfully supported by the ESW cooling tower basin volume.

The above discussion shows that failures of the UHS support systems would be unlikely to trigger an initiating event and can be screened out as negligible contributors to the PRA results and insights.

COL Impact

The Callaway Plant Unit 2 COLA will not be changed as a result of this question.

QUESTION 19-5

Describe how the essential service water emergency makeup system (ESWEMS) pumphouse ventilation system is modeled in the Callaway Plant Unit 2 PRA. If failures of ventilation components are not modeled, provide a quantitative justification for exclusion of these ventilation failures, with reference to failure probabilities, room heat-up assumptions, and operator actions that are possible. (Note that the AREVA NP responses to Questions 19-62 and 19-169 on the U.S. EPR design certification application address design-specific ventilation dependencies.)

Response

The ESWEMS Pumphouse Ventilation System was not explicitly modeled in the Callaway Plant Unit 2 PRA.

This system is described in Section 9.4.15 of the Callaway Plant Unit 2 FSAR. The ESWEMS Ventilation System consists of four trains, one for each train of the ESWEMS. Failure of a component in the ESWEMS Ventilation System train could eventually result in a failure of the ESWEMS train. Combined with an independent failure of the Raw Water Supply System (RWSS), this could lead to the failure of a train of Essential Service Water (ESW), as discussed above in the response to Question 19-4. As discussed in that response, these events are not likely to have significant impact on the plant risk and failures of the UHS support systems can be screened out as negligible contributors to the PRA results and insights.

COL Impact

The Callaway Plant Unit 2 COLA will not be changed as a result of this question.

QUESTION 19-6

The response to Question 19-166 on the U.S. EPR FSAR includes a draft version of Table 19.1-109, which lists assumptions from the PRA. Footnote 2 to the table states that these assumptions will be reevaluated as part of the PRA maintenance and upgrade process and that combined license (COL) item 19.1-9 is provided to confirm that assumptions used in the PRA remain valid for the as-to-be-operated plant. Neither the proposed license condition related to COL item 19.1-9 nor the description of the maintenance and upgrade process in Section 19.1.2.4.1 of the Callaway Plant Unit 2 FSAR refers to this table in the U.S. EPR FSAR. Discuss how this table will be used to ensure that the Callaway Plant Unit 2 PRA reflects the as-to-be-built, as-to-be-operated plant. Revise the FSAR and license condition as appropriate.

Response

The description of how COL item 19.1-9 is addressed in Callaway Plant Unit 2 FSAR Section 19.1.2.2 and in the Part 10 proposed license conditions, will be revised to include a reference to the design certification assumptions found in U.S. EPR FSAR Table 19.1-109.

COL Impact

COLA Part 2, FSAR, and COLA Part 10, Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) and ITAAC Closure, will be revised to summarize the response to this question. The changes are shown in Enclosure 4, Proposed COLA changes associated with RAI Question 19-6.

QUESTION 19-7

Clarify whether the risk metrics resulting from the quantitative screening of external events described in Section 19.1.5 of the Callaway Plant Unit 2 FSAR are outputs of the at-power PRA or the PRA considering all modes of operation. If the at-power PRA was used, provide a similar discussion for external events that occur during shutdown so that the staff can conclude that the impact of external events on total core damage frequency (CDF) and large release frequency (LRF) is not significant.

Response

A supplement will be provided by April 10, 2009 to complete the response to this RAI question.

COL Impact

Later

Enclosure 2

Proposed COLA changes associated with RAI Question 19-1

COLA Part	Chapter	Page No.	Description
2	19	19-6	New bulleted list in FSAR Section 19.1.4.1 providing the site-specific LOOP non-recovery probabilities. Page 19-5 provided for information only.

Industry peer review will be performed for the PRA upgrades, as they are defined above. Appendix A of ASME RA-Sc-2007 (ASME, 2007) provides example revisions to increase clarity on what constitutes an upgrade, versus an update and, therefore, what requires a peer review. When assessing a need for a peer review, consideration will also be given to scope or number of PRA maintenance activities performed. Although individual changes to a PRA model may be considered PRA maintenance activities, the integrated nature of several changes may make a peer review desirable. This is because multiple PRA maintenance activities can, over time, lead to considerable changes in the PRA insights (e.g., relative risk importance of SSCs), and a periodic peer review might be prudent.

Peer reviews will be performed in accordance with Regulatory Guide 1.200 (NRC, 2007a), which endorses NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance" (NEI, 2002), with exceptions. Peer review findings and observations using this process will indicate what improvements are needed to raise the grade given for each PRA technical element. Review findings and observations will be dispositioned based on their importance.

19.1.3 SPECIAL DESIGN/OPERATIONAL FEATURES

No departures or supplements.

19.1.4 SAFETY INSIGHTS FROM THE INTERNAL EVENTS PRA FOR OPERATIONS AT POWER

19.1.4.1 Level 1 Internal Events PRA for Operations at Power

{Two site-specific items are identified as having potential to affect the U.S. EPR PRA model:

- Loss of Offsite Power (LOOP) frequency and duration.
- Balance of plant systems (e.g., Circulating Water System, Normal Heat Sink).

These items were evaluated as follows for potential deviation from the U.S. EPR FSAR.

Loss of Offsite Power

LOOP frequencies used in the U.S. EPR PRA model are consistent with NUREG/CR-6890 guidelines (NRC, 2005a). The LOOP frequency value used in the PRA model is 1.9E-02/yr, based on the generic USA LOOP frequency value of 3.6E-02/yr from NUREG/CR-6890, modified by crediting U.S. EPR full load rejection capability for grid-related events and by excluding consequential LOOP events (consequential LOOP is treated separately in the PRA model).

The base value for LOOP frequency at Callaway Plant Unit 2 site from NUREG/CR-6890 is approximately 3.0E-02/yr. A composite LOOP frequency is calculated by using the U.S. EPR FSAR PRA-generated frequency values for plant- and switchyard-centered LOOP events, and site-specific values for weather and grid-centered LOOP events. This results in a LOOP event frequency (adjusted for consequential LOOP and full load rejection) of approximately 1.7E-02/yr for Callaway Plant Unit 2. This LOOP event frequency is smaller than the value used in the U.S. EPR PRA model (1.9E-02/yr); therefore, the U.S. EPR PRA model is conservative for LOOP event frequency at Callaway Plant Unit 2. In general, given that the generic LOOP frequency for the USA is used in the U.S. EPR PRA, this frequency is likely to be conservative for advanced plants because better plant and switchyard performances are expected. Generic U.S. data are also considered applicable for LOOP recovery values, consequential LOOP values and shutdown LOOP frequency.

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~~A summary of LOOP related conclusions is given below~~The site-specific LOOP nonrecovery probabilities are as follows:

- 1-Hour LOOP nonrecovery probability of 0.516 compared with a U.S. EPR™ value of 0.530.
- 2-Hour LOOP nonrecovery probability of 0.307 compared with a U.S. EPR™ value of 0.318.
- 24-Hour LOOP and nonrecovery probability of 3.95E-05 compared with a U.S. EPR™value of 4.8E-05.

The use of U.S. EPR ™ data for LOOP recovery bounds Callaway Plant Unit 2 site specific values and the difference does not have a significant impact on the PRA results.

For consequential LOOP, there is limited industry data. The U.S. EPR™ FSAR used generic date from NUREG/CR-6890. This data is generic and applicable to Callaway Plant Unit 2.

The U.S. EPR™ shutdown LOOP recovery value is 0.413 and is generic data taken from NUREG/CR-6890. The value is generic and applicable to Callaway Plant Unit 2.

A summary of LOOP related conclusions is given below:

- The U.S. EPR PRA Loss of Offsite Power frequency bounds the Callaway Plant Unit 2 site-specific frequency.
- The U.S. EPR PRA Loss of Offsite Power recovery probabilities bound Callaway Plant Unit 2 site-specific values.
- The U.S. EPR PRA consequential LOOP probabilities do not need to be changed for Callaway Plant Unit 2 because they are not site dependent (they are initiating event dependent).
- The U.S. EPR PRA shutdown LOOP frequency and recovery probabilities are based on generic values and do not need to be changed for Callaway Plant Unit 2.

Site-Specific Balance of Plant Systems

Balance of plant (BOP) systems that are evaluated for potential site specific deviations are the Circulating Water System (CWS), the Closed Cooling Water System (CLCWS), the Auxiliary Cooling Water System (ACWS) and the Normal Heat Sink (NHS).

These site-specific systems were evaluated for differences between the U.S. EPR PRA assumptions and the Callaway Plant Unit 2 site-specific design. It was concluded that the U.S. EPR PRA inputs for the NHS, CWS, CLCWS, and ACWS provide a reasonable and conservative representation of these systems for Callaway Plant Unit 2. This conclusion is based on the following:

- "Loss of Balance of Plant" initiating event is modeled by the fault tree for the BOP support systems. For "Loss of Condenser" and "Loss of Main Feedwater" initiating events the generic initiating event frequencies are used, based on current industry experience (NRC, 2007b). The advanced plants are expected to perform better. Also, the modeling of both loss of main feedwater (generic data) and loss of balance of plant

Enclosure 3

Proposed COLA changes associated with RAI Question 19-2

COLA Part	Chapter	Page No.	Description
2	19	19-7	The Site-Specific Balance of Plant Systems section is revised to include a discussion of the Normal Heat Sink undeveloped event in the U.S. EPR PRA. Page 19-6 is provided for information only.
2	19	19-20	Added a new reference to NUREG/CR-5750.

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~~A summary of LOOP related conclusions is given below~~The site-specific LOOP nonrecovery probabilities are as follows:

- 1-Hour LOOP nonrecovery probability of 0.516 compared with a U.S. EPR™ value of 0.530.
- 2-Hour LOOP nonrecovery probability of 0.307 compared with a U.S. EPR™ value of 0.318.
- 24-Hour LOOP and nonrecovery probability of 3.95E-05 compared with a U.S. EPR™value of 4.8E-05.

The use of U.S. EPR ™ data for LOOP recovery bounds Callaway Plant Unit 2 site specific values and the difference does not have a significant impact on the PRA results.

For consequential LOOP, there is limited industry data. The U.S. EPR™ FSAR used generic date from NUREG/CR-6890. This data is generic and applicable to Callaway Plant Unit 2.

The U.S. EPR™ shutdown LOOP recovery value is 0.413 and is generic data taken from NUREG/CR-6890. The value is generic and applicable to Callaway Plant Unit 2.

A summary of LOOP related conclusions is given below:

- The U.S. EPR PRA Loss of Offsite Power frequency bounds the Callaway Plant Unit 2 site-specific frequency.
- The U.S. EPR PRA Loss of Offsite Power recovery probabilities bound Callaway Plant Unit 2 site-specific values.
- The U.S. EPR PRA consequential LOOP probabilities do not need to be changed for Callaway Plant Unit 2 because they are not site dependent (they are initiating event dependent).
- The U.S. EPR PRA shutdown LOOP frequency and recovery probabilities are based on generic values and do not need to be changed for Callaway Plant Unit 2.

Site-Specific Balance of Plant Systems

Balance of plant (BOP) systems that are evaluated for potential site specific deviations are the Circulating Water System (CWS), the Closed Cooling Water System (CLCWS), the Auxiliary Cooling Water System (ACWS) and the Normal Heat Sink (NHS).

These site-specific systems were evaluated for differences between the U.S. EPR PRA assumptions and the Callaway Plant Unit 2 site-specific design. It was concluded that the U.S. EPR PRA inputs for the NHS, CWS, CLCWS, and ACWS provide a reasonable and conservative representation of these systems for Callaway Plant Unit 2. This conclusion is based on the following:

- "Loss of Balance of Plant" initiating event is modeled by the fault tree for the BOP support systems. For "Loss of Condenser" and "Loss of Main Feedwater" initiating events the generic initiating event frequencies are used, based on current industry experience (NRC, 2007b). The advanced plants are expected to perform better. Also, the modeling of both loss of main feedwater (generic data) and loss of balance of plant (fault tree) initiating events is conservative since the loss of main feedwater contribution is double-counted (due to a loss of the BOP supporting systems).

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- The NHS and the CWS are modeled in the U.S. EPR™ PRA as one undeveloped event, with scope that includes failures of:
    - The NHS
    - The CWS ability to provide cooling to the Main Condenser and to the ACWS System.

This undeveloped event has a failure frequency of 1.0E-02 per year and a failure probability of 2.8E-05 in a 24-hour mission time. These numbers are based on generic industry data from NUREG/CR-6928 (NRC, 2007b) and NUREG/CR-5750 (NRC, 1999). These NUREGs give a frequency of failure of 1.3E-02 per year. The use of 1.0E-02 per year is considered reasonable for the following reasons:

- The value of 1.3E-02 per year included events such as screen plugging, not likely in a closed system, as is used in Callaway Plant Unit 2.
- Loss of Auxiliary Cooling Water events, to which failures of the CWS and NHS contribute, are also included within the Loss of Main Feedwater initiating event and the Loss of Condenser initiating event, multiple-counting some events.

The values used and system characteristics used for the NHS and CWS are generic and/or applicable to Callaway Plant Unit 2.

- In addition, the U.S. EPR PRA unavailability of the NHS is estimated based on the unavailability of the safety UHS that requires operation of one of two cooling fans. This unavailability is expected to bound the unavailability for the Callaway Plant Unit 2 NHS that uses natural draft cooling towers.

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- ~~The CWS is not explicitly modeled in the U.S. EPR PRA. Failures of the CWS are assumed to be enveloped by the failure probability of the NHS.~~ The U.S. EPR PRA model also does not credit the CWS pumps to cool ACWS loads. Callaway Plant Unit 2 has the ability to utilize either the CWS pumps or the ACWS pumps to supply auxiliary cooling water flow to turbine building equipment. Therefore, the ACWS unavailability in the U.S. EPR PRA is expected to bound the unavailability for the Callaway Plant Unit 2 ACWS.
- The Fussell-Vesely importance measures for the evaluated BOP SSCs are low (<0.01%). Based on these importance measures, the applicable U.S. EPR PRA inputs and assumptions would not have a significant impact on the Callaway Plant Unit 2 PRA results and insights.

Conclusions for Level 1 Internal Events PRA for Operations at Power

Based on the above discussion, it is concluded that the U.S. EPR PRA for Level 1 internal events at power is applicable and bounding for the Callaway Plant Unit 2 site. The site and site-specific parameters do not have a significant impact on the PRA results and insights. Therefore, no changes to the U.S. EPR Level 1 internal events PRA are necessary to accommodate specific Callaway Plant Unit 2 site and plant parameters.

19.1.4.2 Level 2 Internal Events PRA for Operations at Power

The discussion presented in Section 19.1.4.1 is also applicable to the U.S. EPR PRA for Level 2 internal events at power because Level 1 and Level 2 event trees are linked together and the

CFR, 2007a. Title 10, Code of Federal Regulations, CFR Part 50.71, Maintenance of Records, Making of Reports, Nuclear Regulatory Commission.

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NEI, 2002. Probabilistic Risk Assessment (PRA) Peer review Process Guidance, NEI 00-02, Revision A3, Nuclear Energy Institute.

NRC, 1977. Protection Against Low-Trajectory Turbine Missiles, Regulatory Guide 1.115, Nuclear Regulatory Commission, July 1977.

NRC, 1991. Procedural and Submittal Guidance for the Individual Plant Examination of External Events, NUREG 1407, Nuclear Regulatory Commission, May 1991.

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Enclosure 4

Proposed COLA changes associated with RAI Question 19-6

COLA Part	Chapter	Page No.	Description
2	19	19-3	Included a reference to U.S. EPR Table 19.1-109 in the COL applicant item response in FSAR Section 19.1.2.2.
10	Appendix A	8	Included a reference to U.S. EPR Table 19.1-109 in the paragraph labeled "COL Item 19.1-9 in Section 19.1.2.2".

19.1.2.2 PRA Level of Detail

The U.S. EPR FSAR includes the following COL Item in Section 19.1.2.2:

> A COL applicant that references the U.S. EPR design certification will review as-designed and as-built information and conduct walk-downs as necessary to confirm that the assumptions used in the PRA, including PRA inputs to RAP and severe accident mitigation design alternatives (SAMDA), remain valid with respect to internal events, internal flooding and fire events (routings and locations of pipe, cable and conduit), and human reliability analyses (HRA) (i.e., development of operating procedures, emergency operating procedures and severe accident management guidelines and training), external events including PRA-based seismic margins, high confidence, low probability of failure (HCLPF) fragilities, and low power shutdown (LPSD) procedures.

This COL Item is addressed as follows:

As-designed and as-built information will be reviewed, and walk-downs will be performed, as necessary, to confirm that the assumptions used in the PRA, including {design certification related PRA assumptions found in U.S. EPR Table 19.1-109 and} PRA inputs to RAP and SAMDA, remain valid with respect to internal events, internal flooding and fire events (routings and locations of pipe, cable and conduit), and HRA (i.e., development of operating procedures, emergency operating procedures and severe accident management guidelines and training), external events including PRA-based seismic margins, HCLPF fragilities, and LPSD procedures. This shall be performed prior to fuel load. eRAI 1839 Q 19-6

19.1.2.3 PRA Technical Adequacy

The U.S. EPR FSAR includes the following COL Item in Section 19.1.2.3:

> A COL applicant that references the U.S. EPR design certification will conduct a peer review of the PRA relative to the ASME PRA Standard prior to use of the PRA to support risk-informed applications or before fuel load.

This COL Item is addressed as follows:

A peer review of the PRA relative to the ASME PRA Standard shall be performed prior to use of the PRA to support risk-informed applications or before initial fuel load.

19.1.2.4 PRA Maintenance and Upgrade

No departures or supplements.

19.1.2.4.1 Description of PRA Maintenance and Upgrade Program

The U.S. EPR FSAR includes the following COL Item in Section 19.1.2.4.1:

> A COL applicant that references the U.S. EPR design certification will describe the applicant's PRA maintenance and upgrade program.

This COL Item is addressed as follows:

The PRA is treated as a living document. The PRA Configuration Control Program maintains (updates) or upgrades the PRA in the manner prescribed by ASME RA-Sc-2007, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (ASME, 2007) and as clarified by Regulatory Guide 1.200 (NRC, 2007a). Thus:

COL Item 18.1-1 in Section 18.1

{AmerenUE} shall execute the NRC approved Human Factors Engineering program as described in U.S. EPR FSAR Section 18.1.

COL Item 18.12-1 in Section 18.12

Prior to initial fuel load, {AmerenUE} shall implement a Human Performance Monitoring Program similar to the one described in Section 18.12 of the U.S. EPR FSAR.

COL Item 19.1-9 in Section 19.1.2.2

As-designed and as-built information shall be reviewed, and walk-downs shall be performed, as necessary, to confirm that the assumptions used in the Probabilistic Risk Assessment (PRA), including {design certification related PRA assumptions found in U.S. EPR FSAR Table 19.1-109 and} PRA inputs to the Reliability Assurance Program and Severe Accident Mitigation Design Alternatives, remain valid with respect to internal events, internal flooding and fire events (routings and locations of pipe, cable and conduit), and Human Reliability Assurance (i.e., development of operating procedures, emergency operating procedures and severe accident management guidelines and training), external events including PRA-based seismic margins, high confidence, low probability of failure fragilities, and low power shutdown procedures. These activities shall be performed prior to initial fuel load.

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COL Item 19.1-4 in Section 19.1.2.3

A peer review of the PRA relative to the ASME PRA Standard shall be performed prior to use of the PRA to support risk-informed applications or before initial fuel load.

COL Item 19.1-5 in Section 19.1.2.4.1

The {Callaway Plant Unit 2} PRA shall be treated as a living document. A PRA Configuration Control Program shall be put in place to maintain (update) or upgrade the PRA, as defined in ASME Standard RA-Sc-2007 and as clarified by Regulatory Guide 1.200.

3. OPERATIONAL PROGRAM IMPLEMENTATION

The provisions of the regulations address implementation milestones for some operational programs. The NRC will use license conditions to ensure implementation for those operational programs whose implementation is not addressed in the regulations. COL application FSAR Table 13.4-1 identifies several programs required by regulations that must be implemented by a milestone to be identified in a license condition.

PROPOSED LICENSE CONDITION:

{AmerenUE} shall implement the programs or portions of programs identified in FSAR Table 13.4-1 on or before the associated milestones in FSAR Table 13.4-1.

4. FIRE PROTECTION PROGRAM REVISIONS

An implementation license condition approved in the Staff Requirements Memorandum (SRM) regarding SECY-05-0197 applies to the fire protection program.

PROPOSED LICENSE CONDITION:

{AmerenUE} shall implement and maintain in effect the provisions of the fire protection program as described in the Final Safety Analysis Report for the facility. The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.