License Renewal Application

Requests for Additional Information (RAI) Responses for Severe Accident Mitigation Alternatives for Callaway Plant, Unit 1

1. Relative to the Level 1 Probabilistic Risk Assessment (PRA):

a. Table F.3-1 appears to be a list of accident sequences contributing to core damage frequency (CDF) as it includes anticipated transient without scram (ATWS), station blackout (SBO), and reactor cooling pump (RCP) seal loss-of-coolant accident (LOCA). Provide a list of initiator event groups and their contribution to total CDF. Separately, please provide the CDF for the ATWS, SBO, and RCP seal LOCA sequences. Also confirm that the SBO frequency includes those following a loss of offsite power (LOSP) as well as those following other transients. If necessary, use an initiating event category of "Other" to ensure the list sums to the total internal events CDF.

Response:

Table 1.a, below, provides a list of Callaway initiating events, and the contribution of each initiating event to the total internal events CDF.

Table 1.a – Contribution of Initiators to Internal Events CDF

Initiating Event (IE)	IE Description	Core Damage Frequency	Percentage of Internal Events CDF
IF	INTERNAL FLOODING	9.14E-06	35.0%
IE-S2	SMALL LOCA INITIATING EVENT FREQUENCY	5.87E-06	22.5%
IE-T1	LOSS OF OFFSITE POWER INITIATING EVENT FREQUENCY	5.58E-06	21.4%
IE-TSG	STEAM GENERATOR TUBE RUPTURE IE FREQUENCY	2.34E-06	9.0%
IE-T3	TURBINE TRIP WITH MAIN FEEDWATER AVAILABLE IE FREQ	1.08E-06	4.1%
IE-S1	INTERMEDIATE LOCA INITIATING EVENT FREQUENCY	3.63E-07	1.4%
IE-TMSO	MAIN STEAMLINE BREAK OUTSIDE CTMT IE FREQUENCY	3.54E-07	1.4%
RV Rupt.	REACTOR VESSEL RUPTURE	3.00E-07	1.1%
IE-S3	VERY SMALL LOCA INITIATING EVENT FREQUENCY	2.07E-07	0.8%
IE-T2	LOSS OF MAIN FEEDWATER IE FREQUENCY	1.88E-07	0.7%
ISLOCA	INTERFACING SYSTEMS LOCA	1.73E-07	0.7%
IE-TC	LOSS OF ALL COMPONENT COOLING WATER IE FREQUENCY	1.20E-07	0.5%
IE-TSW	LOSS OF SERVICE WATER INITIATING EVENT	1.15E-07	0.4%
IE-TFLB	FEEDLINE BREAK UPSTREAM OF CKVS IE FREQUENCY	9.75E-08	0.4%
IE-TDCNK01	LOSS OF VITAL DC BUS NK01 INITIATING EVENT FREQUENCY	4.84E-08	0.2%
IE-A	LARGE LOCA INITIATING EVENT FREQUENCY	4.21E-08	0.2%
IE-TDCNK04	LOSS OF VITAL DC BUS NK04 INITIATING EVENT FREQUENCY	3.13E-08	0.1%
IE-TMSI	MAIN STEAMLINE BREAK INSIDE CTMT IE FREQUENCY	1.54E-08	0.1%
IE-TFLD	FEEDLINE BREAK DOWNSTREAM OF CKVS IE FREQUENCY	1.43E-09	0.0%
	Total Internal Events CDF:	2.61E-05	100%

The CDF for ATWS sequences is 3.14E-7, or 1.2 percent of the internal events CDF. The CDF for station blackout is 4.71E-6, or 18.0 percent of the internal events CDF.

RCP seal LOCAs are explicitly addressed in the Callaway PRA in two ways. First, for non-LOCA transient initiators, a heading is placed early in the event tree to screen for loss of RCP seal cooling. Sequences involving loss of seal cooling are then evaluated / quantified as RCP seal LOCA sequences. The CDF from these RCP seal LOCA sequences is 8.35E-7, or 3.2 percent of the internal events CDF. Second, in certain support system initiator sequences (e.g., loss of all Component Cooling Water, followed by loss of the Normal Charging Pump (NCP)), an RCP seal LOCA is assumed to occur, and is addressed within the given event tree. A third category of RCP seal LOCAs, i.e., seal LOCAs that represent an initiating event, is not explicitly modeled in the Callaway PRA, but can be taken to be included in the Very Small LOCA initiating event category.

ATWS and RCP seal LOCA CDF values, as reported in Table 3-1 of the Environmental Report, have been updated to 3.14E-7 and 8.35E-7 respectively, as stated above.

In the internal events model version used in the SAMA analysis, station blackout sequences are initiated only by the loss of offsite power initiator. (In the upgraded Callaway internal events PRA model, consequential loss of offsite power events, following a loss of main feedwater or reactor trip initiator, are modeled, and can progress to station blackout. In this model, consequential LOOP accounts for approximately 28 percent of the SBO frequency. In addition, consequential LOOP-initiated SBO accounts for 2.5 percent of CDF. Thus, even in the upgraded internal events PRA, station blackout risk is dominated by the loss of offsite power initiator.)

The internal flooding analysis included in PRA Update 4B addresses ATWS and SBO. To simplify the quantification of internal flooding, ATWS sequences are assumed to go directly to core damage. Flooding-initiated sequences that result in SBO would be included via the combination of flood-induced equipment failures and random equipment failures. Note that flood-initiated ATWS and SBO CDF are reported within the “Internal Flooding” CDF category.

b. The internal events CDF is given as 1.66E-05 per year on page F-11. The CDF for the apparent latest revision, Update 4B, is given as 2.61E-05 per year on page F-20. While this difference may be due to exclusion of internal flooding (9.14E-06 in Table 3-4) from the 1.66E-05 value and inclusion of it in the external events multiplier, adding this value for internal floods to the Table 3-1 value yields 2.57E-05, which is close but not equal to the 2.61E-05 value. Provide the basis for the difference in these calculated values and the rationale for the value used in the SAMA analysis. Include discussion of how initiating event contributors were accounted for in each total value.

Response:

The difference in the cited calculated values is due to the inclusion, or not, of the reactor vessel (RV) rupture event in the CDF results. RV rupture is assumed, in the Callaway PRA, to go directly to core damage. A reference frequency of 3E-7 per year is used, in Update 4B, for this event. If the RV rupture frequency is added to the 2.57E-5 value, cited above, the resulting CDF is approximately, i.e., within round-off, the 2.61E-5 value, or the overall Update 4B internal events CDF.

The SAMA analysis essentially uses the 1.66E-5 value, with a multiplier of 4.57 applied to account for external events (refer to the discussion in section 3.1.2.4 of Attachment F). The basis for this approach is that the 1.66E-5 value is the baseline internal events CDF, which is essentially the value perturbed when performing sensitivity analyses in support of the SAMA analysis. All internal initiating event contributors, except for internal flooding and RV rupture, are accounted for in the 1.66E-5 value. Internal flooding is accounted for via use of the 4.57 multiplier. RV rupture is not accounted for in the SAMA analysis. This is acceptable as RV rupture is a singular event that is assumed to go to core damage, and whose probability/frequency would not be directly affected by SAMAs.

c. Provide the truncation value used for each PRA.

Response:

The truncation value used for quantification of the Level 1 internal events PRA used in the SAMA analysis is 1E-12.

d. Provide further discussion of the steps taken to ensure the technical adequacy of the Level 1 PRA subsequent to the 2000 Westinghouse owners group (WOG) peer review. Specifically include in this discussion:

i. Further support for the disposition of peer review facts and observations (F&Os) IE-7 and ST-1 as described in Table 3-8 of the SAMA submittal.

Response:

Further support for the disposition of WOG F&Os IE-7 and ST-1, relative to the SAMA analysis, is provided below.

IE-7:

This F&O provided two comments on the ISLOCA analysis.

The first comment primarily questions whether there could be ISLOCA locations *inside* containment, which could lead to loss of mitigating system (presumed to be ECCS recirc.) function. This comment was investigated, and determined not to be valid for the following reasons:

1. Various documented ISLOCA definitions, e.g., that provided in SR IE-A2, item (d), of the PRA Standard, either explicitly state, or imply, that

ISLOCAs occur *outside* containment.

2. For any LOCA location *inside* the Callaway containment, the water exiting the break will drain to one of the two safety-related containment sumps, and then be available for ECCS recirculation.

Since the F&O comment was determined not to be valid, as described above, there is no associated impact on the SAMA analysis.

The second comment of this F&O is related to the lack of treatment of parametric uncertainty in the ISLOCA analysis. The concern expressed was that this could be important for redundant isolation valves, in an ISLOCA flowpath, that used the same failure rate.

As noted in Table 3-8 of Attachment F, this second F&O comment/issue does not bear negatively on the SAMA analysis. ISLOCA is a very minor contributor to the Callaway internal events CDF (approximately 1%). In addition, the treatment of uncertainty in the SAMA analysis, as described in section 8.2 of Attachment F, is sufficiently conservative so as to bound any impact on the ISLOCA CDF associated with this F&O comment.

<u>ST-1:</u>

This F&O suggests using a reference such as NUREG/CR-5744 to determine the probability of low pressure piping failure upon overpressurization. The F&O states that the reviewers were not familiar with the reference that was used to perform this determination.

This F&O was addressed in an updated ISLOCA analysis performed for PRA Update 5. The methodology in the suggested NUREG was used to determine the probabilities of failure of RHR and SI piping on exposure to RCS pressure. The failure probability of RHR piping on exposure to over-pressure did not change. The failure probability of SI piping increased; however, the overall ISLOCA CDF determined in Update 5 was similar to, and less than, the ISLOCA CDF of Update 4B. The results of the ISLOCA analyses are summarized in the Table 1.d.i, below.

Table 1.d.i – ISLOCA CDF Comparison

Internal Events PRA Version	ISLOCA CDF (yr^{-1})
Update 4B	1.73E-7
Update 5	1.49E-7

Since the overall ISLOCA CDF results are similar, and, in fact, the Update 4B ISLOCA CDF is slightly greater than that for Update 5, WOG F&O ST-1 does not bear negatively on the SAMA analysis.

Note that both the ISLOCA analysis used in PRA Update 4B and the updated ISLOCA analysis of PRA Update 5 include the potential for common cause failure of redundant MOVs and redundant check valves in ISLOCA flow paths.

ii. A description of the findings of the 2006 review against the 2005 revision of the American Society Mechanical Engineers (ASME) PRA standard and the disposition of any deficiencies for the SAMA application. Attachment U of the National Fire Protection Association (NFPA) 805 Licensing Amendment Request (LAR) provides this information relative to the fire risk application. Similar information is needed for the SAMA application including disposition of open findings.

<u>Response:</u>

Table 1.d.ii, below, provides a listing of the gaps between the Callaway internal events Level 1 PRA used for the SAMA analysis and applicable Capability Category II requirements of the 2005 revision of the ASME PRA Standard. The table also provides a disposition of each of the gaps relative to the SAMA application.

Table 1.d.ii

Comparison of Callaway Internal Events Level 1 PRA Used in the SAMA Analysis to Applicable Capability Category II Requirements of the ASME PRA Standard

Supporting Requirement (SR) Not Met at Capability Category II	Associated Finding/ Observation (F/O) No.	F/O Level of Significance	F/O Description	F/O Disposition for SAMA Application
IE-A4	IE-3	C	No documentation of a system-by-system review for IE potential.	System-by-system review has since been performed and documented. No new IEs were identified. Therefore, there is no impact on the SAMA application.
IE-A5	IE-4	C	Non-power events were not evaluated/ addressed in the original IE analysis.	Non-power events have since been addressed. No new IEs were identified. Therefore, there is no impact on the SAMA application.
IE-A7	IE-6	B	Callaway Plant OE for IE precursors was not reviewed when originally identifying IEs.	Callaway precursor OE has since been reviewed. No new IEs were identified. Therefore, there is no impact on the SAMA application.
IE-C1	IE-7	B	The IE frequencies do not have uncertainty bounds assigned.	The SAMA analysis includes a conservative treatment of uncertainty, as discussed in section 8.2 of Attachment F. This treatment of uncertainty bounds the anticipated uncertainties in the IE frequencies used in PRA Update 4B.
IE-C1a	IE-7	B	See discussion for SR IE-C1.	See discussion for SR IE-C1.
IE-C1b	IE-8	B	Certain recovery events, following loss of CCW and loss of SW, are credited, without, in the reviewer's view, sufficient analysis or data.	The credit given (i.e., probabilities used) for the recovery events in question was based on a credible analysis. Should additional analysis determine different recovery probabilities, the resulting uncertainty would be bounded by the overall treatment of uncertainty applied in the SAMA analysis, which is discussed in section 8.2 of Attachment F.

IE-C3	IE-10	C	Certain IE frequencies are not adjusted to account for plant availability.	Adjustment of the IE frequencies in question, to account for plant availability, would result in IE frequencies that are approximately 10 percent lower than those that are not adjusted. Therefore, to the extent that certain IE frequencies in the SAMA analysis do not include this adjustment, the resulting CDF would be conservative.
IE-C9	IE-8	B	See discussion for SR IE-C1b.	See discussion for SR IE-C1b.
IE-C10	IE-12	B	There is no documentation of a comparison of fault tree-generated support system IE frequencies to generic data.	A comparison of all IE frequencies to generic and other plant data was performed and documented for PRA Update 5. No IE frequencies were determined to be outliers as a result of this comparison. Therefore, this gap does not negatively impact the SAMA application.
IE-C12	IE-13	B	Identified gap is related to documentation and age of the Interfacing System LOCA (ISLOCA) analysis.	ISLOCA is only a minor contributor to the Callaway CDF. In addition, the ISLOCA analysis was redone for PRA Update 5 to address gaps to Capability Category II of the ASME PRA Standard. As noted in the response to RAI question 1.d.i, the Update 5 ISLOCA CDF is similar to, and slightly lower than, the Update 4B ISLOCA CDF. Therefore, this gap does not negatively impact the SAMA application.
IE-C13	IE-7	B	See discussion for SR IE-C1.	See discussion for SR IE-C1.
IE-D1	IE-14	C	Finding is that, while IE documentation is reasonably complete, it is not conducive to performing updates or peer reviews, primarily because the IE documentation resides in a relatively large number of documents. Finding was categorized by review team as a documentation issue.	This gap was deemed by the review team to be a documentation issue. Adequate documentation of the IE analysis does exist. This finding does not negatively impact the SAMA application.
IE-D2	IE-14	C	See discussion for SR IE-D1.	See discussion for SR IE-D1.
IE-D3	IE-14	C	See discussion for SR IE-D1.	See information for SR IE-D1.

AS-A11	AS-2	B	This finding was based on there being event tree transfer sequences that were quantified with .OCL (i.e., sequence quantification) files that were generated manually, and not with a specific event tree. This was deemed to introduce the possibility of errors, although none were found.	The transfer sequences have been extensively reviewed, and no issues have been identified. Therefore, this F/O does not negatively impact the SAMA application.
AS-B1	AS-1, AS-3, AS-5, AS-7	B, B, C, B, respectively	The 4 cited F/Os essentially question whether the impact of certain IEs, on certain mitigation functions credited in event trees (ETs), was correctly captured.	A sensitivity evaluation performed for the previously-approved one-time ESW Completion Time (CT) extension application determined that correction of F/Os AS-1, -3 and -7 would result in only a 1% increase in the Update 4 baseline CDF. (Note also that certain F/O-implied issues were investigated, and could not be verified.) F/O AS-5 suggests re-evaluation of the switchgear room cooling requirements for SBO conditions. It has since been determined that switchgear room cooling is not required for any initiator. Use of the Update 4B model, which requires room cooling, for the SAMA application, would be conservative. Based on the above discussion, the gap to Capability Category II of SR AS-B1 would not negatively impact the SAMA application.
AS-B2	AS-1, AS-3, AS-5, AS-7	B, B, C, B, respectively	See discussion for SR AS-B1.	See discussion for SR AS-B1.
AS-B6	AS-4, AS-5	B, C, respectively	AS-4 cites the need to update the RCP seal LOCA model. AS-5 recommends re-evaluating the room cooling requirement for switchgear rooms during SBO.	AS-5 is discussed above for SR AS-B1. Regarding AS-4, the seal LOCA model used in Update 4B is based on an older-vintage Westinghouse Owners' Group study. A previous sensitivity study indicated that baseline CDF would increase by only 1.5 percent when seal LOCA model-related parameters were varied (i.e., increased). Thus, the AS-4 finding would not negatively impact the SAMA application.

SC-B5	SC-2	C	No documentation of a check of the reasonableness of success criteria.	The review team deemed this to be a documentation issue. The Callaway Plant success criteria are similar to the Wolf Creek success criteria. (Wolf Creek is essentially the same plant design as Callaway.) During development of the original PRA model, for the IPE, periodic comparisons were made of the Callaway and Wolf Creek PRAs, including comparisons of success criteria. The Callaway Plant success criteria are also comparable to success criteria for other, similar plants. In addition, for PRA Update 5, the various event tree success criteria were validated using MAAP 4.0.7. No significant changes were identified. Based on the above discussion, this finding does not negatively impact the SAMA application.
SC-C1	SC-1	C	Success criteria are not documented in a single place.	The review team deemed this to be a documentation issue. Thus, this gap to Capability Category II of the SR does not negatively impact the SAMA application.
SC-C2	SC-1	C	See discussion for SR SC-C1.	See discussion for SR SC-C1.
SC-C3	SC-1	C	See discussion for SR SC-C1.	See discussion for SR SC-C1.
SY-A7	SY-1	B	Two issues were identified: (1) the dependency of Main Feedwater on instrument air (IA) needs to be included in the model and (2) the applicability of data used for undeveloped events for loss of IA and failure of actuation signals needs to be verified.	A sensitivity analysis was performed to address these issues for a previous application. The analysis resulted in only a 0.59 percent increase in the Update 4 baseline CDF. Therefore, this finding does not negatively impact the SAMA application.
SY-A22	IE-8	B	See discussion for SR IE-C1b.	See discussion for SR IE-C1b.

SY-B1	SY-2	B	CCFs are not modeled for battery chargers or breakers. In addition, the quantification of CCF probabilities should be updated.	Battery charger and breaker independent failure events have low Fussel-Vesely importances in PRA Update 4. The F-V importance of CCFs of battery chargers and breakers would also be expected to be relatively low, if these failure modes were modeled. As a sensitivity analysis, battery charger common cause failure events were added to the Update 4 PRA model, and the model was re-quantified. There was no discernable change in core damage frequency. In fact, all battery charger CCF events added to the model were truncated from the core damage cutset results. (A truncation value of approximately seven (7) orders of magnitude below the baseline CDF was used in the PRA quantification.) A separate sensitivity analysis was performed in which all existing CCF probabilities were increased by 10 percent. The PRA Update 4 baseline CDF increased by only 3.54 percent. Based on the above discussion, this finding does not negatively impact the SAMA application.
SY-B3	SY-2	B	See discussion for SR SY-B1.	See discussion for SR SY-B1.
HR-D3	HR-1	C	Suggestion for addition of a "ground rule" statement to HRA documentation.	The review team deemed this finding to be a documentation issue. As such, this finding does not negatively impact the SAMA application.
HR-G6	HR-2	C	A reasonableness check of HEPs was performed, but not documented.	The review team deemed this finding to be a documentation issue. As such, this finding does not negatively impact the SAMA application.
HR-I3	HR-3	C	No documentation of key sources of uncertainty associated with the HRA.	The review team deemed this finding to be a documentation issue. As such, this finding does not negatively impact the SAMA application.

DA-B1	DA-2	B	Only Capability Category I is met with respect to SR DA-B1. (Components were not grouped according to characteristics of their usage.)	The data update task performed for PRA Update 5 grouped components by component type **and** characteristics of their usage (in order to meet Capability Category II of this SR). The resulting groupings had populations that were similar to the groupings that are the subject of this finding. Therefore, this finding would not negatively impact the SAMA application.
DA-C2	DA-2	B	See discussion for SR DA-B1.	See discussion for SR DA-B1.
DA-C6	DA-1	C	Documentation of certain data collection is lacking. (The reviewer noted, however, that the correct information appears to have been collected.)	The review team deemed this finding to be a documentation issue. As such, and since no actual errors were noted, this finding does not negatively impact the SAMA application.
DA-C7	DA-1	C	See discussion for SR DA-C6.	See discussion for SR DA-C6.
DA-C8	DA-1	C	See discussion for SR DA-C6.	See discussion for SR DA-C6.
DA-C9	DA-1	C	See discussion for SR DA-C6.	See discussion for SR DA-C6.
DA-C14	IE-8	B	See discussion for SR IE-C1b.	See discussion for SR IE-C1b.
DA-D2	DA-3	C	No justification is provided for the use of engineering judgment to determine the probabilities of HYDRAULIC-SYSFAIL, STR-FS, STR-FR basic events.	The review team deemed this finding to be a documentation issue. A sensitivity analysis was performed in which the probabilities of the HYDRAULICSYSFAIL and all STR basic events were increased by a factor of 2. The PRA Update 4 baseline CDF increased by only 0.03 percent. Based on the above discussion, this finding does not negatively impact the SAMA application.

IF-C2a	IF-5	B	The SR requires that operator response to floods be based on flood area and flood sources. The Callaway Plant IF analysis, through Update 4B, treats operator response in a generic sense.	The Callaway internal flooding (IF) analysis was redone for PRA Update 5 to update the analysis and address gaps to Capability Category II of the Standard. The Update 5 IF CDF is 6.21E-6, or about two-thirds of the Update 4B IF CDF. Due to the method for inclusion of the IF CDF in the SAMA analysis, i.e., including the IF contribution to CDF in the external events multiplier (refer to section 3.1.2.4 of Attachment F), use of the larger Update 4B IF CDF is conservative. Therefore, the various findings related to the IF analysis do not negatively impact the SAMA application.
IF-C6	IF-3	C	Current Callaway Plant flood area screening credits operator intervention for floods that take >30 mins. Criteria for Capability Category II are not explicitly addressed.	Refer to F/O disposition discussion for SR IF-C2a.
IF-C8	IF-3	C	See discussion for SR IF-C6.	Refer to F/O disposition discussion for SR IF-C2a.
IF-D5	IF-1	C	The flood initiator frequencies are based on generic pipe break frequencies. No plant-specific experience was considered in the determination of flood initiator frequencies.	Refer to F/O disposition discussion for SR IF-C2a.
IF-D5a	IF-1	C	See discussion for SR IF-D5.	Refer to F/O disposition discussion for SR IF-C2a.
IF-E3a	IF-2	B	Standard specifies a CDF screening criterion of 1E-9; existing Callaway Plant IF analysis used 1E-6.	Refer to F/O disposition discussion for SR IF-C2a.
IF-E5	IF-4	B	HEPs for operator intervention and mitigation are not based on HRA, as required by the Standard.	Refer to F/O disposition discussion for SR IF-C2a.
IF-E5a	IF-4	B	See discussion for SR IF-E5.	Refer to F/O disposition discussion for SR IF-C2a.
QU-A2b	QU-1	B	Current PRA does not include an uncertainty calculation accounting for the “state-of-knowledge” correlation.	Uncertainty is included in the SAMA analysis via the methodology described in section 8.2 of Attachment F. Therefore, this finding does not negatively impact the SAMA application.

QU-D4	QU-5	C	No documentation of a review of non-significant sequences or cutsets.	The review team deemed this gap to be a documentation issue. As such, it does not negatively impact the SAMA application. It is noted that all accident sequences have been reviewed via QR (qualified review) of the event trees, regardless of their frequency. In addition, non-significant cutsets have been reviewed from time-to-time, e.g., following a PRA update, or pursuant to applications.
QU-E3	QU-1	B	See discussion for SR QU-A2b.	See discussion for SR QU-A2b.
QU-F1	QU-8	C	Recommendation to integrate all pieces of the internal events analysis into one quantification process.	When using Update 4B, separate steps are, in fact, required to quantify internal events, internal flooding and/or ISLOCA. However, as long as the contribution to risk from each of these sources can be determined (including via the use of multipliers), this finding does not negatively impact the SAMA application.
QU-F2	QU-9	B	Some of the typical QU documentation items cited in the Standard do not exist for Update 4.	The review team deemed this gap to be a documentation issue. As such, it does not negatively impact the SAMA application.
QU-F4	QU-10	B	Key assumptions and key sources of uncertainty "are not addressed in a coherent manner in the documentation."	The review team deemed this finding to be a documentation issue. As such, it does not negatively impact the SAMA application. In addition, uncertainty is addressed in the SAMA analysis.
QU-F5	QU-12	C	No documentation exists of limitations in the quantification process that would impact applications, per the Standard requirements.	The review team deemed this finding to be a documentation issue. As such, it does not negatively impact the SAMA application.
QU-F6	QU-11	B	Definitions of significant cutset and significant accident sequence, used in PRA Update 4, differ from those of the Standard. Justification of the alternative definitions used was not justified, as required by the Standard.	The review team deemed this finding to be a documentation issue. As such, it does not negatively impact the SAMA application.

<u>Notes on Table 1d.ii</u>:

1. SR numbers based on the 2005 version of the ASME PRA Standard.
2. Definitions of F/O Significance Levels:

   A. Extremely important and necessary to address to assure the technical adequacy of the PRA or the quality of the PRA or the quality of the update process.
   (There was no "A" level F/Os.)

   B. Important and necessary to address, but may be deferred until the next PRA update.

   C. Marginal importance, but considered desirable to maintain maximum flexibility in PRA applications and consistency in the industry.

   D. Editorial or minor technical item, left to the discretion of the host utility.
   (No "D" level F/Os were written.)

iii. Discussion of findings from a Human Reliability Analysis (HRA) focused scope peer review. We understand from Attachment U of the NFPA 805 LAR submittal that the internal events HRA modeling has been revised and undergone a focused scope peer review. This peer review is not discussed in the SAMA submittal. Discuss the scope of this review and disposition of open findings.

Response:

As noted in Table 3-3 of Attachment F, the Fourth Update to the internal events PRA, i.e., "PRA Update 4", implemented an updated human reliability analysis (HRA) for risk-significant human failure events (HFEs). One of the purposes of this HRA update was to address open HRA-related Westinghouse Owners' Group (WOG) peer review findings, as the WOG peer review deemed the HRA element to be Grade 2 (refer to page F-32). The updated HRA is also used in PRA Update 4B, the model used in the SAMA analysis.

In 2011, an independent, focused-scope peer review of the updated Callaway internal events HRA was performed. The purpose of the review was to examine updated elements of the HRA relative to the findings from the WOG peer review and with respect to the current (i.e., 2009) version of the Standard for probabilistic risk assessment (PRA). The review concluded that the detailed analyses of risk-significant HFEs appeared to reflect appropriate methods properly implemented. Some observations were made in the course of the review, but were judged to be relatively minor, meriting consideration in further updates to the HRA. The focused-scope peer review of the HRA update did not generate any findings per se. Some observations were made. All of the observations were judged by the reviewer to be relatively minor, meriting consideration in future updates to the HRA.

e. As a result of the NRC review of the Callaway NFPA 805 submittal, NRC staff has requested the results of sensitivity analyses to show the impact of potentially unacceptable modeling approaches (see PRA RAI-08 on influence weighting factors and PRA RAI-09 on control power transformer credit). Please provide the sensitivity of these unacceptable fire PRA modeling approaches on the calculated fire CDF. If this NFPA 805 sensitivity is not bounded by the SAMA 95th sensitivity analysis provide the impact of this higher fire CDF on the SAMA evaluation.

Response:

PRA-RAI-08a from the NFPA-805 LAR involves the usage of compartment weighting factors for transient fire apportionment. The method used in the Callaway Fire PRA was shown to be within the bounds of the approved NRC methods. The approved NRC method requires that at least one of the 3 weighting factors is set to 1.0 and thus, the total weighting factor is not less than 1.0. The

Callaway Fire PRA used a combined minimum weighting factor of 1.15 and therefore is within the bounds of the approved NRC methods. No sensitivity study was performed on this issue for the RAI.

PRA-RAI-09a concerns the use of data from Table 10.1 in NUREG/CR-6850. The Callaway PRA used values from Table 10.1 for probability of spurious operation for components with a control power transformer in the control circuit. New fire test data shows minimal effect of CPT's. The RAI asked for a recalculation of CDF using the spurious operation probabilities from Table 10.2 of NUREG/CR-6850 (i.e., for thermoset cables without CPT). When using the Table 10.2 values, the point estimate of fire CDF increased from 2.04E-5 to 2.67E-5, an increase of 6.3E-6.

The internal events CDF at Callaway is 1.66E-5. The 95th percentile value for internal events CDF is 3.50E-5. This provides an uncertainty factor of 2.11 for point estimate to 95th. The total external events CDF (due to Fire, Internal flood, high winds, seismic) is 5.91E-5. If the uncertainty factor of 2.11 from the internal events is applied to the external events, the 95th percentile of the external events CDF is 1.25E-4. The CDF increase of 6.3E-6 from the CPT sensitivity study performed for PRA-RAI-9a is well within the 95th value for external CDF.

f. Provide the freeze date for PRA Update 4B and include in your response whether there have been any changes to the plant, either physical or procedural, since that date that could have a significant impact on the results. If there have been significant changes that represent an increase in risk, provide the impact of those changes on the SAMA evaluation.

Response:

The purpose of PRA Update 4B was to incorporate the Alternate Emergency Power System (AEPS) into the Callaway internal events PRA model. Update 4B was completed in February, 2011, and reflected the as-built, as-operated plant at that time. In addition, Callaway has various programs in place to screen plant hardware and procedure changes for their impact on the Callaway PRA. There have been no physical or procedural changes to the plant, since the completion of Update 4B, that would have a significant impact on the PRA results, or the SAMA analysis.

g. Table 3-2 on page F-13 includes the basic event TORNADO-T1-EVENT with a risk reduction worth (RRW) of 1.031. Please explain the basis for this event being included in the internal events PRA.

Response:

The Callaway internal events PRA includes credit for an Alternate Emergency Power System (AEPS). This system is comprised of an off-site facility with both a connection to a Cooperative power line and four (4) 2 MWe diesel-generators. Following certain loss of offsite power events, with subsequent failure of the on-site, safety-related emergency diesel-generators, AC power can be manually aligned, from either the Cooperative power line or the 2 MWe diesel-generators, to one of the 4160 VAC safety-related electrical busses. However, should the loss of offsite power have been caused by a tornado, it is assumed that AEPS is not available. Thus, the “TORNADO-T1-EVENT” basic event, which represents the conditional probability that the LOOP event was caused by a tornado, is used in the internal events PRA to fail the AEPS.

The probability of this event is determined as follows:

*P(tornado) / P(LOOP) = 5E-4 / 1.6E-2 = **3.125E-2**,*

where P(tornado) is taken from page 2.3-7 of the Callaway FSAR Site Addendum. (Note that this same tornado frequency is used in the estimation of tornado-initiated CDF of section F.3.1.2.3, and is referred to in the response to RAI 3.b.)

2. <u>Relative to the Level 2 PRA:</u>

a. The 5th bullet in Section F.3.2 indicates that the sequences that contribute to large early release frequency (LERF) were determined based on source term calculations using Modular Accident Analysis Program (MAAP) 4.0.7. Please provide the basis for the source terms for the other release categories.

<u>Response:</u>

All source terms were determined using MAAP 4.0.7, with the results from MAAP used to categorize the sequences as LERF or non-LERF per the ASME PRA Standard.

b. The last paragraph in Section F.3.2 states “There were no changes to major modeling assumptions, containment event tree structure, accident progression, source term calculations or other Level 2 attributes, used in the individual plant examination (IPE) Level 2 analysis, when developing the initial and updated models.” Justify this statement in light of the many apparent changes discussed previously in this section and in the disposition of the Level 2 Peer Review Facts/Observations (F&Os) in Table F.3-8, or provide a discussion of the changes.

<u>Response:</u>

This statement is referring to the “individual plant examination (IPE) Level 2 analysis”, “Initial LERF Model” in 2000 (first row of the table) and “Updated LERF Model” in 2002 (second row of the table). The changes discussed previously in that section and in the disposition of the Level 2 Peer Review F&Os in Table F.3-8 refer to changes between the IPE “Updated LERF Model” in 2002 (second row of the table) and the “Updated full Level 2 Model” in 2011 (last row of the table).

c. Provide a description of the containment event tree (CET) or trees used in the level 2 analysis including a listing and description of the CET nodes. Include a description of how phenomenological events and containment system failures are addressed in the CET.

Response:

Section 2.2 of the Callaway Level 2 Analysis is provided in Enclosure 2, and describes the containment event tree structure.

d. Section F.3.4 identifies eight release categories. Provide further information on each release category including: category definitions and their bases; how the CET end states are assigned to release categories; a description of the sequences that are the major contributors to each release category; the basis for the selection of MAAP case used for each release category; and a description of the MAAP cases used. Also, if the source terms for each release category are not bounding, then provide justification of how the impact of higher source term sequences are accounted for in determining the benefit of potential SAMAs, or provide a sensitivity analysis using bounding case source terms.

Response:

Sections 2.5 and 3.1 of the Callaway Level 2 Analysis are provided in Enclosure 2. Sections 2.5.1 and 2.5.3 provide the release category definitions and bases. CET endstate assignments are included in Section 2.5.2 (also see CET in previous RAI). Major contributors to each release category are included in Section 3.1.1 of the Level 2 report.

A unique MAAP case was created and run for each release category in Table 3-1 of the Level 2 report (e.g., MAAP run LERF-CI for case LERF-CI). Additional MAAP runs for LERF categories with more than one significant contributor were also performed to ensure the representative sequence was valid (e.g., LERF-CI, LERF-CF).

The categories were selected based on the types of sequences that Callaway produces. Failures or bypasses of containment lead to generally early releases. Otherwise, a long time occurs until a later release due to containment overpressure or basemat melt through. Intermediate time sequences do not generally occur, and so no such category was needed.

e. Provide a discussion of the steps taken to insure the technical adequacy of the Level 2 PRA. Include, as part of your response, identification of peer reviews, gap analyses, or other reviews that were performed for the Level 2 PRA and when these reviews were performed.

Response:

Several steps were taken to ensure the technical adequacy of the Level 2 PRA. Within the technical staff of the contractor that performed the Level 2 PRA (ERIN Engineering & Research Inc.), an internal review was performed as evidenced by the signature sheet on the Level 2 PRA. Prior to finalization, the Level 2 PRA and report were also reviewed by Ameren/Callaway staff.

In addition, Appendix E of the Level 2 PRA report provides a self-assessment of the 2011 Level 2 PRA against the LE supporting requirements from the ASME PRA Standard. This roadmap identifies how the Level 2 PRA addresses the Category II LE supporting requirements. No gaps related to the Level 2 analysis were identified. The Level 2 roadmap is included as Enclosure 4 and provided with permission from *ERIN Engineering and Research, Inc.*

f. Table 3-6 gives the importance results for LERF and Table 3-7 gives the importance results for Late Release. Since there are five LERF release categories and two late release categories, please explain which release categories were included in these assessments, (i.e. all or just the largest contributor).

Response:

The importance results for LERF and Late Releases use the general LERF and LATE release categories which combine the detailed LERF and LATE release categories. That is, LERF is the combination of the five LERF subcategories LERF-IS, LERF-CI, LERF-CF, LERF-SG, & LERF-ITR, while LATE is the combination of the two LATE subcategories LATE-BMT and LATE-COP.

3. **Relative to External Events**

a. Section F.3.1.2.2 states the following: "For the individual plant examination external events (IPEEE), Callaway used the Electric Power Research Institute (EPRI) seismic margins analysis (SMA) method. This analysis was transmitted to NRC in the IPEEE submittal. The latest estimate of the Callaway seismic contribution to CDF is 5.00E-6/yr."

A SMA does not normally include an estimate of seismic CDF. Please explain the source and basis for the 5.00E-6/yr value.

Response:

The 5.00E-6/yr seismic CDF estimate was not developed directly from the IPEEE SMA. This estimate was derived using engineering insights from the IPEEE and more current studies (e.g. the NRC's Generic Issue 199 risk assessment report) for the sole purpose of developing the total external events multiplier used in the SAMA analysis.

There were no significant seismic vulnerabilities identified at the Review Level Earthquake acceleration analyzed by the IPEEE; thus the plant design was determined to be seismically robust. The Generic Issue 199 risk assessment calculated a seismic CDF of 2.3E-6 using the weakest link model for Callaway. However, that risk assessment was a relatively generic assessment. To account for modeling uncertainties Engineering judgment was used and the calculated value was doubled and rounded up for use in developing the seismic contribution to the total external events multiplier.

b. Section F.3.1.2.3 states that the risk for tornado events is 2.5E-05/yr. and this is considered a contributor to the external events initiator group for calculating the external events multiplier.

   i. Provide the basis for computation of this value. Include in this description consideration of buildings that are not tornado hardened and systems that could be failed by the tornado.

Response:

Callaway does not have a high winds PRA and thus an engineering estimate of high winds CDF contribution was developed. This estimate was derived using engineering insights from the IPEEE and the internal events PRA for the sole purpose of developing the total external events multiplier used in the SAMA analysis.

The most relevant value for tornado frequency calculated in the FSAR Site Addendum, section 2.3.1.2.6.1, is 5E-4/yr. Due to the robust building design basis for tornado loading, described in FSAR section 3.3.2.1, it is virtually certain that no safety related equipment credited in the PRA would be damaged by any tornado. However, when combined with random failures of safety related equipment, it is still assumed that a tornado could contribute to CDF by impacting offsite power and other PRA credited equipment not protected by hardened structures. It is conservatively assumed that 50 percent of potential tornados would be of sufficient strength and size to cause wide spread damage to unprotected equipment. Assuming that a tornado disables the following unprotected equipment: normal and alternate sources of offsite power, non-safety related service water and the non-safety related auxiliary feedwater pump; the conditional core damage probability (CCDP) is on the order of 1E-3. If it is assumed that the ultimate heat sink (safety related water source) is also disabled, the CCDP could approach 1.0. Given that there is a range of CCDPs associated with different combinations of wind damaged equipment and varying combinations of random equipment failures, a CCDP of 0.1 was chosen as a representative value for the convolution of CCDPs from potential damage states. Thus, 5E-4/yr * 0.5 * 0.1 results in a conservatively estimated 2.5E-5/yr CDF from tornados.

ii. Identify SAMAs to mitigate the contribution this makes to the total CDF.

Response**:**

SAMA 15 is related to tornado impacts; however this contribution is included in the internal events loss of offsite power initiator and is not considered an external event. There are no other SAMA items directly related to tornado impacts. New SAMAs intended to protect the equipment referenced in the response above would be of such a scale that they would be physically infeasible from an engineering perspective and prohibitively expensive from a SAMA cost benefit perspective.

4. <u>Relative to the Level 3 analysis</u>

a. Tables F.3-9 and 3-10 provide the year 2044 population distribution used in the MELCOR Accident Consequence Code System, Version 2 (MACCS2) analysis. Since the SECPOP2000 code was utilized to develop initial residential population estimates for each spatial element within the 50 mile region based on year 2000 census data, provide the year 2000 population distribution (Table 2.6-1 provides only a partial breakdown).

<u>Response:</u>

The year 2000 population distribution is provided in Tables 4a-1 and 4a-2 below, in a format similar to that of Tables F.3-9 and F.3-10. This year 2000 population includes permanent residents (developed from the SECPOP2000 code) and population associated with transients and special populations for the 0-10 mile region (developed from the Callaway evacuation time estimate study). (See response to RAI 4c regarding the inclusion of special populations). The year 2000 population distribution provided in Tables 4a-1 and 4a-2 was used as the basis for the population projection to year 2044.

Table 4a-1
Year 2000 Population Distribution Within a 10-Mile Radius

Sector	0-1 mile	1-2 miles	2-3 miles	3-4 miles	4-5 miles	5-10 miles	10 mile Total
N	5	5	50	136	55	202	453
NNE	6	19	50	50	69	262	456
NE	6	5	0	16	29	52	108
ENE	6	7	0	0	0	95	108
E	6	5	0	0	77	109	197
ESE	16	5	2	11	34	140	208
SE	6	5	0	46	64	169	290
SSE	5	5	4	0	0	179	193
S	0	0	3	2	0	976	981
SSW	0	51	0	50	10	88	199
SW	0	0	0	0	74	1461	1535
WSW	0	0	0	0	28	548	576
W	0	131	0	0	0	583	714
WNW	0	56	84	83	102	852	1177
NW	0	0	1	15	5	789	810
NNW	0	0	26	24	7	455	512
Total	56	294	220	433	554	6960	8517

Table 4a-2
Year 2000 Population Distribution Within a 50-Mile Radius

Sector	0-10 miles	10-20 miles	20-30 miles	30-40 miles	40-50 miles	50 mile Total
N	453	803	6498	1356	1901	11011
NNE	456	564	2533	2036	5610	11199
NE	108	861	3790	1333	2285	8377
ENE	108	524	3120	3608	17555	24915
E	197	1517	1154	12724	18022	33614
ESE	208	3148	2168	6700	34651	46875
SE	290	769	1352	5248	7069	14728
SSE	193	420	928	6785	4519	12845
S	981	1934	1624	3297	2592	10428
SSW	199	2291	2794	1996	2664	9944
SW	1535	1468	14347	4912	3642	25904
WSW	576	6035	43896	12195	7166	69868
W	714	2481	5767	2454	3572	14988
WNW	1177	5989	15341	95759	3525	121791
NW	810	9801	2338	8352	5217	26518
NNW	512	2400	8287	6253	1534	18986
Total	8517	41005	115937	175008	121524	461991

b. Section F.3.4.1 identifies that the population was projected to year 2044 using county growth estimates. Please describe how the county growth rates were applied (e.g., county weighted per sector, or State average uniformly applied across all sectors). In addition, if sectors or counties were projected to have negative growth, describe how they were treated.

Response:

To obtain the year 2044 population projection, individual county growth rates were applied at the polar coordinate grid element level such that adjacent grid elements may use different growth rates if different county data applies. Some grid elements include land from multiple counties. In such cases, a weighted growth rate was developed for those grid elements based on the fraction of land in that grid element associated with each county.

The county growth rates were developed based on Missouri Office of Administration (MOA) projected population data. MOA provides projected population data for each county in 5 year increments, from year 2000 to 2030. County growth rates were calculated using the MOA population data from year 2000 (census data) and year 2025 (MOA projection), not year 2030 as implied in Section F.3.4.1. The year 2025 was chosen in lieu of 2030 based upon the year 2025 being closer to the mid-point of the total required projection period (from year 2000 to 2044). Originally, consideration was given to projecting the population out to year 2050, and use of the 2000-2025 period would allow applying the same growth factor for each grid element successively to project to 2050. Projection to year 2050 was subsequently judged to be overly conservative, and year 2044 (end of license) was selected for use. The MOA-based county growth rates were applied, as applicable, to each grid element as discussed above, to calculate a year 2025 projected population distribution. This process was then repeated to project from year 2025 to year 2044, using the year 2000-2025 growth factors with an adjustment factor of 0.76 applied (i.e., 19yr/25yr) to represent the shorter projection period associated with years 2025 to 2044.

Two counties were projected to have negative growth in the year 2000 to 2025 time period, Howard and Montgomery. Zero growth (rather than negative growth) was conservatively assumed for these two counties for the Callaway population projection.

It is noted that projecting the population to year 2044 (the last year of the Callaway license renewal term) rather than "a year within the second half of the period of extended operation" (NEI 05-01, Section 3.4.1), which could be as early as 2034, provides conservatism.

c. Section F.3.4.1 identifies that transient population data was included within the 10-mile radius. Provide the year 2000 transient population and identify whether the transient population was scaled to the year 2044. Briefly discuss how the year 2000 transient population was included within the 10-mile radius. If transient population was not addressed, provide the impact of accounting for transient population on the SAMA evaluation.

<u>Response:</u>

Transient population data was included in the 10-mile radius prior to the population projection such that transient population data was scaled to year 2044 using grid element growth factors developed for the permanent resident population data. The transient population data was obtained from Figure 6a of the Callaway evacuation time estimate (ETE) study. It is noted that the ETE also identified "special facility" population data associated with individuals in institutions such as hospitals, nursing homes, schools, and jails. This special facility population data could include individuals (e.g., hospital patients) not included in census based residential population data. Therefore, this special facility data (from ETE Figure 7a) was included with the identified transient population data prior to the population projection such that special facility data was also scaled to year 2044 using grid element growth factors developed for the permanent resident population data. Inclusion of special facility data may result in some conservatism because some special facility populations such as nursing homes and jails may be included in permanent resident data.

The ETE transient and special facility population data (Figures 6a and 7a, respectively) are provided for each of the 16 sector directions for radial intervals of 0-to-2 miles, 2-to-5 miles, and 5-to-10 miles. For the 0-to-2 miles and 2-to-5 miles intervals, the population was divided evenly between the grid elements used in the MACCS2 analysis. If the population could not be divided evenly, the additional population was added to the grid element(s) closest to the site.

Table 4c-1 provides the year 2000 transient (including special facility) population distribution. The special facility population is specifically provided per the table notes.

Table 4c-1
Year 2000 Transient (Including Special Facility) Population Distribution Within a 10-Mile Radius

Sector	0-1 mile	1-2 miles	2-3 miles	3-4 miles	4-5 miles	5-10 miles	10 mile Total
N	5	5	50	50	50	0	160
NNE	6	5	50	50	50	0	161
NE	6	5	0	0	0	0	11
ENE	6	5	0	0	0	0	11
E	6	5	0	0	0	0	11
ESE	5	5	0	0	0	42	52
SE	6	5	0	0	0	42	53
SSE	5	5	0	0	0	42	52
S	0	0	0	0	0	345(1)	345
SSW	0	0	0	0	0	58	58
SW	0	0	0	0	0	1155(2)	1155
WSW	0	0	0	0	0	0	0
W	0	0	0	0	0	0	0
WNW	0	0	84	83	83	0	250
NW	0	0	0	0	0	0	0
NNW	0	0	0	0	0	0	0
Total	45	40	184	183	183	1684	2319

Notes:

1) Includes special facility population of 287 persons.

2) Includes special facility population of 1097 persons.

d. Section F.3.4.2 identifies that some generic economic data was used from NUREG-1150, and scaled using the consumer price index (CPI) to May 2010. Please provide the effective cost escalation factor applied.

Response:

An effective cost escalation factor of 2.0 was applied to escalate generic economic data used from NUREG-1150 (estimated to date from 1986) to May 2010. The cost escalation factor was derived from the CPI, as follows:

1986 Annual CPI = 109.6

May 2010 CPI = 218.178

Cost escalation factor = 218.178 / 109.6 = 1.991, rounded up to 2.0

e. Three sector population and economic estimator (SECPOP) 2000 code errors have been publicized, specifically: (1) incorrect column formatting of the output file, (2) incorrect 1997 economic database file end character resulting in the selection of data from wrong counties, and (3) gaps in the 1997 economic database numbering scheme resulting in the selection of data from wrong counties. Address whether these errors were corrected in the Callaway analysis. If they were not corrected, then provide a revised cost-benefit evaluation of each SAMA with the errors corrected.

Response:

All three SECPOP2000 code errors were accounted for in the original submittal such that the errors did not impact the Callaway analysis. Error #1 (incorrect column formatting of the SECPOP2000 output file which can lead to incorrect column formatting of the SITE input file) was accounted for by ensuring that the column alignment of the final SITE input file was correct (e.g., manually adjusting the column formatting as needed). Errors #2 and #3 both involve the 1997 economic database associated with SECPOP2000. Section F.3.4.2 notes that the 1997 economic database was not used because of the age of the data. Instead, more recent data (e.g., 2007 Census of Agriculture, 2008 Bureau of Economic Analysis, 2010 Bureau of Labor Statistics) was obtained and the calculation approach documented in NUREG/CR-6525 (SECP2000) was used to develop regional economic data inputs. This was done in an electronic spreadsheet rather than using SECPOP2000. Because the 1997 economic database associated with SECPOP2000 was not used, the associated errors (#2 and #3) were avoided.

f. The emergency response sensitivity shows a +7 percent change for slower evacuation and a +2.4 percent change for delayed evacuation. Is the higher impact for evacuation speed due to unsheltered travel and/or exposure to "higher" initial dose releases versus early sheltering and lower delayed releases?

Response:

Sensitivity to emergency response inputs (i.e., delay time to evacuation, evacuation speed) was evaluated by means of sensitivity analysis. Section 3.4.4 summarizes that the first sensitivity case evaluated the impact of increased delay time before evacuation begins (i.e., vehicles begin moving in the 10-mile region). For this sensitivity case, the base delay time of 105 minutes was doubled to 210 minutes. The second sensitivity case evaluated the impact of a reduced evacuation speed. For this second sensitivity case, the evacuation speed was halved from 2.14 m/s (4.8 mph) to 1.07 m/s (2.4 mph). Section 3.4.4. specifies that the increased delay time case and the decreased evacuation speed case shows a dose increase of about 2.4% and 7%, respectively. Review of these dose impact values indicates that they are incorrect. The 2.4% and 7% reported values were based on release frequencies that were subsequently updated, but the text of Section 3.4.4 was not updated appropriately.

Additionally, while reviewing the Callaway MACCS2 modeling to respond to the RAIs, an error was found involving the evacuation zone distance. The MACCS2 model used to develop the results for the Attachment F SAMA analysis incorrectly modeled evacuation of individuals within 20 miles of the Callaway plant rather than within 10 miles, as intended. The model was corrected and a MACCS2 sensitivity case was run to determine the impact of the error on the base case analysis. The impact was found to be very small. The population dose risk increased from 4.60 person-rem/yr to 4.65 person-rem/yr (an approximately 1% increase). The cost risk did not change. To provide a perspective on this change, the impact on the SAMA analysis is less than the change associated with using a different year of weather data. Table 4f-1 provides the results of the sensitivity case with the corrected evacuation modeling for comparison to Table 3-14 (which was based on evacuation of individuals within 20 miles). The ingestion dose results (Table 3-15) are not impacted by this evacuation correction because ingestion dose is accumulated over the long term.

Table 4f-1
Base Case Evacuation Region Corrected Dose and Cost Results
(0-50 Mile Radius from Callaway Site)

Release Category	Frequency (per yr)	Dose (p-rem)	Dose Risk (p-rem/yr)	Cost ($)	Cost Risk ($/yr)
LERF-IS	1.73E-07	1.91E+06	3.30E-01	8.22E+09	1.42E+03
LERF-CI	1.66E-10	7.81E+05	1.30E-04	4.80E+09	7.97E-01
LERF-CF	1.13E-08	8.44E+05	9.54E-03	5.49E+09	6.20E+01
LERF-SG	2.33E-06	9.35E+05	2.18E+00	4.92E+09	1.15E+04
LERF-ITR	2.17E-07	1.22E+06	2.65E-01	8.01E+09	1.74E+03
LATE-BMT	2.55E-06	3.92E+04	1.00E-01	4.91E+07	1.25E+02
LATE-COP	3.19E-06	5.45E+05	1.74E+00	1.86E+09	5.93E+03
INTACT	8.08E-06	2.87E+03	2.32E-02	1.25E+06	1.01E+01
Total	1.66E-05	--	4.65E+00	--	2.08E+04

When the distance within which people were evacuated was corrected from 20 miles to 10 miles, the impact on different release categories varied. For most release categories, the population dose increased as would generally be expected because fewer people were evacuating (i.e., those in the 10-to-20 mile region were no longer evacuating.) Those individuals who were no longer evacuating would generally be expected to receive additional dose. For two release categories (i.e., LERF-IS and LERF-ITR) the dose decreased slightly. Review of the release timing for these two release categories indicates that people in the 10-to-20 mile region were just beginning their evacuation movement at approximately the time the first release plume was entering the 10-to-20 mile region. (The average wind speed in the 2008 meteorological data is approximately 7.3 mph, which roughly corresponds to a time of 1.4 hours before the plume reaches 10 miles. Wind speeds associated with specific weather sequences used in a MACCS2 run will vary.) People who are evacuating are modeled to have less shielding than those who are not evacuating. Therefore, by eliminating the evacuation in the 10-to-20 mile region, these individuals received less dose for these releases. The cost impacts associated with correcting the evacuation region distance was negligible.

The sensitivity cases for the evacuation delay time (i.e., 210 minutes) and slower evacuation speed (i.e., 1.07 m/s (2.4 mph)) were repeated using the corrected evacuation zone distance. The dose results are presented in Table

4f-2 for evacuation delay and Table 4f-3 for evacuation speed. The dose results of Tables 4f-2 and 4f-3 may be compared to those of Table 4f-1.

Table 4f-2 demonstrates that increasing the delay time from 105 minutes to 210 minutes increases the dose risk by approximately 2.7% (compared to the corrected base case analysis). Table 4f-3 demonstrates that decreasing the evacuation speed from 1.14 m/s to 1.07 m/s increases the dose risk by approximately 6% (compared to the corrected base case analysis).

Per Table 4f-2, the increased delay time results in decreased dose for two release categories (i.e., LERF-IS and LERF-ITR). Both of these releases involve short duration initial plumes (i.e., less than 2 hours) with relatively high release fractions (i.e., > 0.1 CsI) that occur relatively rapidly compared to the time of GE declaration. As a result, the increased delay time affords individuals increased shielding (e.g., at their residences) while the first plume passes over before their evacuation movement begins (when less shielding is provided by their vehicles). In comparison, the slower evacuation movement case (Table 4f-3 results) does not afford this additional shielding effect, and therefore, a slower evacuation speed has a larger impact (for the parameter values utilized).

Table 4f-2
Increased Delay Time Sensitivity (Evacuation Region Corrected) Dose Results
(0-50 Mile Radius from Callaway Site)

Release Category	Frequency (per yr)	Dose (p-rem)	Dose Risk (p-rem/yr)	Dose Risk Change from Base Case (%)
LERF-IS	1.73E-07	1.77E+06	3.06E-01	-7.3%
LERF-CI	1.66E-10	7.92E+05	1.31E-04	1.4%
LERF-CF	1.13E-08	8.53E+05	9.64E-03	1.1%
LERF-SG	2.33E-06	1.00E+06	2.33E+00	7.0%
LERF-ITR	2.17E-07	1.19E+06	2.58E-01	-2.5%
LATE-BMT	2.55E-06	3.92E+04	1.00E-01	0.0%
LATE-COP	3.19E-06	5.46E+05	1.74E+00	0.2%
INTACT	8.08E-06	2.87E+03	2.32E-02	0.0%
Total	1.66E-05	--	4.77E+00	2.7%

Table 4f-3
Decreased Evacuation Speed Sensitivity (Evacuation Region Corrected) Dose Results
(0-50 Mile Radius from Callaway Site)

Release Category	Frequency (per yr)	Dose (p-rem)	Dose Risk (p-rem/yr)	Dose Risk Change from Base Case (%)
LERF-IS	1.73E-07	2.01E+06	3.48E-01	5.2%
LERF-CI	1.66E-10	7.95E+05	1.32E-04	1.8%
LERF-CF	1.13E-08	8.52E+05	9.63E-03	0.9%
LERF-SG	2.33E-06	1.04E+06	2.42E+00	11.2%
LERF-ITR	2.17E-07	1.29E+06	2.80E-01	5.7%
LATE-BMT	2.55E-06	3.92E+04	1.00E-01	0.0%
LATE-COP	3.19E-06	5.46E+05	1.74E+00	0.2%
INTACT	8.08E-06	2.88E+03	2.33E-02	0.3%
Total	1.66E-05	--	4.93E+00	6.0%

g. Provide the MAAP and MACCS2 (if different than MAAP) radioisotope grouping and identify the release time for early versus late release.

Response:

MAAP 4.0.7 (used for Callaway) uses 12 radioisotope groups as follows:

Group #	Description
1	Noble (Xe, Kr) and Inert aerosols
2	CsI, RbI
3	TeO_2
4	SrO
5	MoO_2, RuO_2, TcO_2
6	CsOH, RbOH
7	BaO
8	La_2O_3, Pr_2O_3, Nd_2O_3, Sm_2O_3, Y_2O_3, ZrO_2, NbO_2
9	CeO_2, NpO_2, PuO_2
10	Sb
11	Te_2
12	UO_2

MACCS2 uses nine radioisotope groups as follows:

Group #	Description
1	Xe, Kr
2	I
3	Cs, Rb
4	Te, Sb
5	Sr
6	Ru, Co, Mo, Tc, Rh
7	La, Y, Zr, Nb, Pr, Nd, Am, Cm,
8	Ce, Np, Pu
9	Ba

The 12 MAAP groups were mapped to the nine MACCS2 groups as follows:

MACC2 Group #	MAAP Group #
1	1
2	2
3	6
4	3, 10, 11
5	4
6	5
7	8
8	9, 12
9	7

For cases where multiple MAAP groups were assigned to a MACCS2 group, the largest release fraction of the MAAP groups was used for the MACCS2 group.

In the Callaway Level 2 PRA, the distinction between early and late releases includes consideration of the associated plant specific source term such that the distinction is between LERF and non-LERF. In general releases that occur within a few hours of a declaration of a general emergency (GE) and have a CsI cumulative release fraction approaching or above 0.1 are defined as LERF. Releases that have a lower CsI release fraction or tend to occur later, such as due to containment overpressure or basemat meltthrough, are defined as non-LERF releases. In the Callaway release category naming scheme, these non-LERF releases are identified as “LATE-“ because they have a significant delay prior to significant release.

Attachment F Table 3-13 identifies the time of GE declaration and the times of release of individual plumes for each release category used in the MACCS2 analysis, based on the MAAP plant specific analysis. As a result, the MACCS2 modeling is not dependent upon a specific definition of early versus late releases.

h. Identify the specific reference for the Callaway Evacuation Study. In your response, please discuss whether and how the evacuation time was adjusted for the difference in population between year 2045 and the year of the referenced evacuation time estimate study. If the evacuation time was not adjusted for the difference in population between year 2045 and the year of the referenced evacuation time estimate study, briefly discuss the potential impact to the SAMA evaluation. Identify whether the emergency planning zone (EPZ) was treated as a single evacuation zone.

Response:

The specific reference for the Callaway Evacuation Study is as follows:

Evacuation Time Estimate for the Callaway Nuclear Plant Emergency Planning Zone, July 2002, Appendix G of Radiological Emergency Response Plan (RERP) Revision 36.

No adjustment was made to the estimated evacuation time to account for potential differences between the year 2044 population (the year of the population projection for the SAMA analysis) and the year of the time estimate study (2002) for the MACCS2 basecase analysis. It is generally assumed that increases in population will be accompanied by increases in infrastructure (e.g., road widening) during this lengthy time period such that the evacuation speed will not significantly decrease. The base case evacuation speed assumed for the analysis is 2.14 m/s (4.8 mph).

Sensitivity to evacuation speed was evaluated by means of sensitivity analysis, as discussed in response to RAI 4f. The results of Table 4f-3 may be compared to those of Table 4f-1 and demonstrate that reducing the evacuation speed by one half results in an increase of the population dose risk from 4.65 p-rem/yr to 4.93 p-rem/yr, an increase of approximately 6%. This dose increase is relatively modest and demonstrates that a relatively significant decrease in the evacuation speed will not have a significant impact on the SAMA dose risk.

Table 4h-2
Slower Evacuation and Evacuation Region Corrected Dose and Cost Results
(0-50 Mile Radius from Callaway Site)

Release Category	Frequency (per yr)	Dose (p-rem)	Dose Risk (p-rem/yr)	Cost ($)	Cost Risk ($/yr)
LERF-IS	1.73E-07	2.01E+06	3.48E-01	8.22E+09	1.42E+03
LERF-CI	1.66E-10	7.95E+05	1.32E-04	4.80E+09	7.97E-01
LERF-CF	1.13E-08	8.52E+05	9.63E-03	5.49E+09	6.20E+01
LERF-SG	2.33E-06	1.04E+06	2.42E+00	4.92E+09	1.15E+04
LERF-ITR	2.17E-07	1.29E+06	2.80E-01	8.01E+09	1.74E+03
LATE-BMT	2.55E-06	3.92E+04	1.00E-01	4.91E+07	1.25E+02
LATE-COP	3.19E-06	5.46E+05	1.74E+00	1.86E+09	5.93E+03
INTACT	8.08E-06	2.88E+03	2.33E-02	1.25E+06	1.01E+01
Total	1.66E-05	--	4.93E+00	--	2.08E+04

For evacuation, 95% of the population is assumed to evacuate the 10 mile radius portion of the EPZ, radially away from the site. This 95% evacuation is modeled as a single evacuation zone. Five percent of the population is assumed to not participate in the evacuation.

i. Section F.3.4.5 indicates that the year 2008 meteorological data was more conservative than years 2007 and 2009. Describe the basis for this assertion and briefly quantify the relative conservatism. In addition, please identify the meteorological tower heights (i.e. potential range of measurement elevations) for the onsite meteorology station and for the station at the Prairie Fork Conservation.

Response:

[Note: In the ERIN MACCS2 report, Table 4.2-1 provides the results of the meteorological sensitivities (see cases CALMET07, CALMET09). For these cases the change in population dose risk and cost risk ranged from -4% to 0%. These results formed the basis for the characterization of year 2008 met data providing conservative results. Following finalization of the ERIN report, the Level 2 release category frequencies were apparently updated (i.e., the frequencies in Table F.3-14 are different than in ERIN Report Table 4.2-1). The changes in the release frequencies result in year 2008 data being less conservative, and non-conservative for cost risk for 2007. The RAI response addresses the 2007 data using different justification however.]

Table 4i-1 provides the frequency-weighted 50-mile population dose risk and cost risk results for each of the years of weather data evaluated, calculated using the MACCS2 model used for the results documented in Attachment F.

Table 4i-1
Attachment F Callaway Meteorological Sensitivity Results

Met Data Year	50-mile Pop Dose Risk (p-rem/yr)	Delta from 2008 Basecase	50-mile Cost Risk ($/yr)	Delta from 2008 Basecase
2008 (base case)	4.60	--	2.08E+04	--
2007	4.58	-0.4%	2.09E+04	0.5%
2009	4.60	0%	2.02E+04	-2.9%

For the 2009 sensitivity case, the 50-mile population dose risk and cost risk exhibit no change (dose risk) or a slight decrease (cost risk). For the 2007 sensitivity case, the dose risk showed a slight decrease while the cost risk showed a slight increase.

RAI response 4f identifies an error discovered in the MACCS2 evacuation modeling that results in small changes to the risk metrics. When the evacuation modeling is corrected, the results of the meteorological sensitivities cases are as shown in Table 4i-2.

Table 4i-2
Callaway Meteorological Sensitivity Results for Corrected Evacuation

Met Data Year	50-mile Pop Dose Risk (p-rem/yr)	Delta from 2008 Basecase	50-mile Cost Risk ($/yr)	Delta from 2008 Basecase
2008 (base case)	4.65	--	2.08E+04	--
2007	4.61	-0.9%	2.09E+04	0.5%
2009	4.62	-0.6%	2.02E+04	-2.9%

Use of the year 2008 meteorological data is slightly conservative in terms of dose risk and cost risk compared to year 2009 meteorological data. For year 2007, the dose risk is conservative but the cost risk is slightly non-conservative. For reasons discussed below, year 2008 data were preferred over year 2007 data.

In addition to Callaway on-site meteorological data, additional weather data was obtained from nearby Prairie Fork Conservation Station due to incomplete Callaway on-site precipitation data. Use of data from the Prairie Fork Conservation Station was limited to precipitation data only (i.e., not wind speed or wind direction) for portions of year 2007 only (i.e., the 2008 and 2009 meteorological data sets used in the MACCS2 analysis did not incorporate any data from the Prairie Fork Conservation Station). Use of precipitation data from Prairie Fork Conservation Station was warranted for 2007 due to significant precipitation data voids (approximately 23% for year 2007) associated with Callaway on-site precipitation gage malfunctions and system upgrades. The inclusion of off-site precipitation data for year 2007 to fill significant data voids was a determining factor in not selecting the 2007 meteorological data set for the base case MACCS2 analysis.

The Callaway on-site meteorological data is collected at heights of 10m, 60m and 90m. The Prairie Fork Conservation Station precipitation data is collected near ground level (at approximately 0.8m)

j. Provide the values and associated assumptions made about the following MACCS2 input parameters: rainfall, mixing heights, building wake effects, plume release energy, land fraction, region index, watershed index, growing season, fraction of farmland, and shielding and protection factors.

<u>Response:</u>

<u>Rainfall</u>

Site specific precipitation data was used for the Callaway MACCS2 analysis. Site specific precipitation data for year 2007 was supplemented by data from the Prairie Fork Conservation Station. (See response to RAI 4i for further discussion on the use of Prairie Fork Conservation Station precipitation data). The total (i.e., annual) precipitation included in each MACCS2 meteorological file is as follows:

Year	Total Precipitation
2007	27.7 inches
2008 (base case)	53.1 inches
2009	60.3 inches

Mixing Heights

The atmospheric mixing height values included in the MACCS2 meteorological input files were based on Holzworth (*Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States, EPA, January 1972).* This is a copyrighted reference and is therefore not supplied with this response. The following mixing height values were used in the Callaway MACCS2 analysis:

Time	Winter	Spring	Summer	Autumn
Morning	460m	500m	350m	350m
Afternoon	890m	1600m	1690m	1400m

Building Wake Effects

In MACCS2 the initial size of the plume is influenced by the building wake which is dependent on the width (W) and height (H) of the building. For the Callaway MACCS2 analysis, values for the initial plume standard deviation parameters sigma-y (SIGYINIT) and sigma-z (SIGXINIT) were calculated using plant specific containment dimensions and the following relationships documented in the MACCS2 User's Guide:

Sigma-y = W / 4.3

Sigma-z = H / 2.15

The Callaway containment width is approximately 45.3 m, and the containment height is approximately 63.6 m above grade level (from Callaway drawing M-2G029). Therefore, the initial plume parameters used are as follows:

Sigma-y = 45.3 m / 4.3 = 10.5 m

Sigma-z = 63.6 m / 2.15 = 29.6 m

These values were assumed for each of the plume segments associated with each release category.

The building height (MACCS2 parameter WEBUILDH) was a straight input of the Containment Building's height of 63.6m.

Plume Release Energy

For the base case MACCS2 analysis zero plume energy was assumed. A sensitivity case was performed to examine the impact of plume heat content. The sensitivity case assumed 1E+07 watts for each plume of each release, except for the intact containment case in which the assumption of zero plume heat was maintained. When plume heat content is included, the dose risk decreases approximately 2.1% and the cost risk increases approximately 1.7%. This sensitivity case demonstrates that the assumption related to plume release energy has a very small impact on the MACCS2 results. (These results are based on using the corrected evacuation region MACCS2 model discussed in response to RAI 4f.)

Land Fraction

The fraction of each spatial element that is land (as opposed to water) was visually estimated using maps and images of the 50-mile region. Table 4j-1 shows the land fractions used in the MACCS2 model.

Table 4j-1
Callaway Spatial Element Land Fractions

Sector	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
N	0.95	0.99	0.96	0.95	0.95	0.99	0.99	0.98	0.98	0.95
NNE	0.95	0.99	0.98	0.99	0.94	0.99	0.99	0.99	0.98	0.99
NE	0.90	0.99	0.99	0.99	0.96	0.99	0.99	0.99	0.98	0.98
ENE	0.97	0.99	0.99	0.98	0.99	0.99	0.98	0.98	0.98	0.98
E	0.99	0.99	0.99	0.97	0.98	0.99	0.98	0.97	0.98	0.98
ESE	0.99	0.99	0.99	0.99	0.99	0.95	0.95	0.97	0.95	0.95
SE	0.98	0.99	0.99	0.99	0.90	0.95	0.98	0.98	0.99	0.98
SSE	0.95	0.99	0.99	0.99	0.80	0.98	0.98	0.98	0.98	0.98
S	0.97	0.99	0.99	0.99	0.98	0.96	0.99	0.97	0.97	0.97
SSW	0.97	0.99	0.99	0.99	0.98	0.95	0.98	0.97	0.99	0.98
SW	0.97	0.99	0.99	0.97	0.96	0.98	0.96	0.94	0.97	0.97
W	0.98	0.99	0.99	0.96	0.99	0.99	0.99	0.95	0.98	0.98
WNW	0.97	0.99	0.99	0.96	0.99	0.99	0.99	0.99	0.96	0.98
NW	0.99	0.99	0.99	0.98	0.97	0.99	0.99	0.99	0.98	0.97
NNW	0.98	0.99	0.97	0.97	0.98	0.99	0.99	0.99	0.99	0.99
N	0.98	0.99	0.97	0.95	0.96	0.99	0.99	0.99	0.99	0.98

Region Index

A total of 63 regions are defined for the MACCS2 model for the 50-mile region surrounding the Callaway plant. Spatial elements of the 50-mile polar coordinate grid within the same county have the same index value. Spatial elements involving multiple counties have unique index values.

Watershed Index

All spatial elements within the 50-mile region surrounding the Callaway plant are designated as river systems. Per NUREG/CR-4551 the designation of lake is only used for very large bodies of water, such as Lake Michigan, which may serve as drinking water sources. The lakes around the Callaway site are smaller and are expected to behave like river systems.

Growing Season

Growing season and crop share data are required to be included in the MACCS2 SITE input file. Table 4j-2 provides the growing season and crop share data included in Callaway SITE file. These data are consistent with the values in the MACCS2 Users Guide. These data, however, are only implemented in MACCS2 when the old MACCS2 food model is used. For Callaway, the new COMIDA2 food model was used. Therefore, these growing season values have no impact on the Callaway MACCS2 results.

Table 4j-2
Callaway Crop Growing Season Inputs

Crop	Calendar Growth Start	Calendar Growth End	Crop Share
Pasture	90	270	0.41
Stored Forage	150	240	0.13
Grains	150	240	0.21
Green Leafy Vegetables	150	240	0.002
Other Food Crops	150	240	0.004
Legumes and Sees	150	240	0.15
Roots and Tubers	150	240	0.003

Fraction of Farmland

The fraction of farmland in each spatial element was estimated using county farm fraction data obtained from the 2007 Census of Agriculture, multiplied by the percentage of county in each spatial element (i.e., a land area weighting was applied to the county data). Table 4j-3 shows the farmland fractions used in the Callaway MACCS2 model corresponding to the region index.

Table 4j-3
Callaway MACCS2 Region Farm Fraction Input

Region #	Description	Farm Fraction		Region #	Description	Farm Fraction
1	EXCLUSION	0.61		33	SE-40	0.50
2	CALLAWAY	0.61		34	OSAGE	0.77
3	N-20	0.91		35	SSE-10	0.72
4	N-30	0.96		36	SSE-20	0.67
5	N-40	0.75		37	SSE-30	0.64
6	NNE-10	0.64		38	SSE-40	0.48
7	NNE-20	0.79		39	S-30	0.72
8	NNE-30	0.93		40	S-40	0.69
9	NNE-40	0.86		41	SSW-05	0.73
10	NE-5	0.63		42	SSW-30	0.74
11	NE-10	0.71		43	SSW-40	0.70
12	MONTGOM	0.72		44	SW-5	0.64
13	NE-30	0.75		45	SW-10	0.67
14	NE-40	0.76		46	SW-20	0.74
15	ENE-05	0.68		47	SW-30	0.72
16	ENE-20	0.69		48	SW-40	0.66
17	ENE-30	0.59		49	WSW-20	0.64
18	ENE-40	0.62		50	WSW-30	0.76

Region #	Description	Farm Fraction		Region #	Description	Farm Fraction
19	E-05	0.69		51	WSW-40	0.85
20	E-10	0.69		52	W-20	0.59
21	E-20	0.53		53	W-30	0.76
22	WARREN	0.54		54	W-40	0.91
23	E-40	0.48		55	WNW-20	0.59
24	ESE-05	0.68		56	BOONE	0.59
25	ESE-10	0.67		57	WNW-40	0.78
26	ESE-20	0.56		58	NW-20	0.64
27	ESE-30	0.52		59	NW-30	0.61
28	ESE-40	0.50		60	NW-40	0.73
29	SE-05	0.73		61	NNW-20	0.84
30	SE-10	0.65		62	NNW-30	0.89
31	SE-20	0.62		63	NNW-40	0.83
32	SE-30	0.54		--	--	--

Shielding and Protection Factors

Shielding and protection factors used in the Callaway MACCS2 model are presented in Table 4j-4. Shielding and protection factors are taken from NUREG-1150 (NUREG/CR-4551). Shelter values for cloud shine and ground shine are based on those used in NUREG-1150 for Sequoyah. Sequoyah values were chosen because they represent reasonable mid-range values (for the five sites evaluated in NUREG-1150) and because the Sequoyah site is one of the closest NUREG-1150 sites to Callaway, such that similar conditions are expected.

Table 4j-4
Callaway Shielding and Protection Factors

Description	Evacuees	Normal Activity	Shelter
Cloud Shielding Factor	1	0.75	0.65
Ground Shielding Factor	0.5	0.33	0.20
Protection Factor for Inhalation	1	0.41	0.33
Skin Protection Factor	1	0.41	0.33

k. Table 3-15 provides ingestion doses. Identify the model(s) and version used and the critical input parameters used to produce these results.

Response:

Food ingestion is modeled using the COMIDA2 food ingestion model (provided with MACCS2), consistent with Sample Problem A (i.e., the Sample Problem A COMIDA2 binary output file, SAMP_A.bin, was used as input to the MACCS2 calculation).

Water ingestion data inputs (e.g., NUMWPI, NAMWPI, WSHFRI, WSHRTA, WINGF) included in the CHRONC file are consistent with Sample Problem A, based on NUREG/CR-4551 values.

Food ingestion dose limit data inputs are based on 1998 FDA Guidance (Accidental Radioactive Contamination of Human Food and Animal Feeds: Recommendations for State and Local Agencies) that superseded the 1982 FDA Guidance incorporated in Sample Problem A. The values used for Callaway are as follows:

Dose Limit Variable	Effective (Sv)	Thyroid (Sv)
DOSEMILK001	0.0025	0.025
DOSEOTHR001	0.0025	0.025
DOSELONG001	0.005	0.050

It is noted that the ingestion doses identified in Table 3-15 are not affected by the evacuation modeling change identified in the response to RAI 4f.

5. <u>Relative to the selection and screening of Phase I SAMA candidates</u>

a. Table F.5-1 shows that while 6 out of the 171 SAMA candidates identified are plant specific SAMAs identified from plant-specific risk insights, it appears that the fire PRA for the recently submitted NFPA 805 LAR was not used as a source to generate plant-specific risk insights. Table F.3-4 shows that the external event contribution to total CDF is greater (e.g., fire CDF is 2.0E-5/yr) than the internal events contribution (i.e., internal CDF is 1.7E-5/yr). Provide identification and evaluation of SAMAs based on plant specific insights from the post-transition fire PRA. Include, as part of this identification, consideration of fire PRA importance analysis, the dominant risk fire areas and associated sequences, and the risk of modifications that Callaway has committed to. Also, describe how this information was used to identify SAMA candidates and evaluate any resulting SAMA candidates not already evaluated. When evaluating the impact of additional SAMAs consider whether the fire related SAMA can have additional benefit from non-fire contributors.

Response:

The results of the NFPA 805 Fire PRA were not available at the time the SAMA analysis was performed and therefore could not be used to identify and evaluate any SAMA candidates. The following summarizes the requested information.

Fire Areas Contributing >1% of Fire CDF and/or Fire LERF

The following table shows individual fire areas that contribute at least 1% of total Fire CDF or Fire LERF, excluding the Multi-compartment (MCA) fire risk. The cumulative MCA risk is below 2E-7/yr, which is 1% of the total fire risk.

Table 5-2: Risk Significant, Individual Fire Areas

Fire Area	CDF (1/yr)	LERF (1/yr)	Total CDF Contribution	Total LERF Contribution
TB-1	6.54E-06	1.36E-07	32.4%	34.3%
A-21	5.13E-06	6.12E-08	25.4%	15.4%
C-10	1.72E-06	3.85E-08	8.5%	9.7%
C-22	1.35E-06	2.58E-08	6.7%	6.5%
YD-1	1.03E-06	2.18E-08	5.1%	5.5%
C-9	9.31E-07	2.10E-08	4.6%	5.3%
C-27	7.83E-07	2.06E-08	3.9%	5.2%
C-21	5.10E-07	4.17E-08	2.5%	10.5%
A-29	2.63E-07	2.29E-10	1.3%	0.1%

Table 5-2: Risk Significant, Individual Fire Areas				
Fire Area	**CDF (1/yr)**	**LERF (1/yr)**	**Total CDF Contribution**	**Total LERF Contribution**
A-17	2.50E-07	2.29E-09	1.2%	0.6%
RB-1	2.36E-07	1.93E-09	1.2%	0.5%
A-13	2.04E-07	1.78E-10	1.0%	0.0%
C-24	1.30E-07	6.15E-09	0.6%	1.5%

The top 8 fire areas accounting for 89% of CDF and 92.4% of LERF are summarized below. The fire frequency in each area is comprised of many individual fire scenarios. There are approximately 1,300 fire scenarios in the Callaway fire PRA. The top 16 individual fire scenarios are discussed after the area discussion. The top 16 individual fire scenarios account for 58% of CDF and 59% of LERF.

Risk Significant Fire Areas

Fire Area **TB-1** has the highest CDF because it is the largest fire area and has the highest fire ignition frequency of any fire area. These fires do not generally fail any safety related equipment. The principle fire related failures are Offsite power and the non-safety service water system. Although there are generally two safety related trains to support shutdown after a turbine building fire, the random failures of these systems can lead to core damage. The average CCDP of turbine building fires is 2.3E-4, which is indicative of the failure of two safety related cooling water systems.

Fire Area A-21 has the opposite risk characteristics of TB-1. There are only 5 ignition sources in this area, but one of the ignition sources has two risk significant fire scenarios contributing to Fire CDF. Note that fire suppression is not credited in A-21.

Fire Area C-10 is an electrical switchgear room. The risk significant fires in this room will fail Offsite power, and train related safety equipment. There are switchgear fires in this room which have a high relative initiating event frequency and are postulated to be damage intensive. Extensive fire modeling was performed in this area, especially for the main buss. Automatic suppression is credited to reduce the risk of fires where possible.

Fire Area C-22 is a cable spreading room, which has a limited total ignition frequency of transient fires only. Although frequency of fires is low (automatic fire suppression is also credited), when fires do occur they can cause significant damage due to the cable content of the area. The risk profile of this room is the same as C-10.

The **Yard (YD-1**) is a CDF contributor because it causes Loss of Offsite Power, which in turn causes loss of the non-safety service water system. A simplified fire modeling approach was used in the YARD in which ten sub-areas or “analysis areas” were analyzed as whole-area burnouts. As such, within each analysis area the assumed fire damage was conservative, which, along with a relatively high ignition frequency, causes moderate risk. The dominant fires involve the transformer fires.

Fire Area C-9 is an electrical switchgear room. The risk profile of this switchgear room is similar to C-10, but pertains to the opposite train.

Fire Area C-27 is the Main Control Room (MCR). Risk significant fires in the control room involve main control board fires which fail significant safety related equipment of both trains (as well as MCR evacuation).

Fire Area C-21 is a cable spreading room. The risk profile of this cable spreading room is similar to C-22, but pertains to the opposite train.

<u>Fire Scenarios Contributing Top 95% of Fire CDF and LERF</u>

The quantification of individual areas report supporting the NFPA-805 submittal contains tables showing all quantified fire scenarios (including fire-modeled and whole-area burnup scenarios) that contribute to the Fire CDF and LERF.

Fire scenarios contributing > 1% of Fire CDF are described in detail below:

<u>Scenario 1501-1A</u>

Notable failures include MDAFP "B" via suction valves spurious close (SC), CCW "B" via EGHV16/54 SC and EFHV52 SC, EDG "B" via EFHV60 spurious open (SO), and all 4 RCP seal injection valves (8351A/B/C/D) SC. The fire damage leaves the plant running on Train "A" with no seal injection available from the NCP. Cutsets are dominated by spurious fire-induced failures of CCW "A", spurious closure of any one RCP seal injection valve (leading to seal LOCA), and failure to initiate recirc after successful injection.

Scenario 1501-1

This scenario is dominated by an RCP seal LOCA of 176 gal per minute in one or more pumps with successful ECCS injection, but failures in the ECCS recirculation mode due to a) human errors, b) spurious opening of EGTV0030, or c) spurious closure of EFHV0052. The loss of seal cooling is caused by spurious closure of the BBHV8351 valves [fire damage] and spurious closure of the CCW thermal barrier cooling isolation valves due to false signal from EGFT0062. Charging pumps and CCW pump are available, but blockage in the seal injection line and the CCW thermal barrier line isolate seal cooling to all RCPs. After 13 minutes, a 176 gpm LOCA is postulated to occur in each pump.

Scenario YD-SXFR

This scenario involves a large transformer fire in the YARD. It fails all offsite power from the main switchyard to PA01 and PA02. Offsite power is available to NB01 and NB02 from the Alternate Emergency Power System (AEPS). The dominant pathway to core damage is a loss of RCP seal cooling and a failure to provide RCS makeup in response. AFW is available throughout the sequence. Contributors to risk are failures of both trains of ESW. The non-safety service water is unavailable due to LOSP. Loss of all ESW causes loss of all ECCS, CCW and the charging pumps. Non-safety charging pump is unavailable due to LOOP. Loss of seal cooling leads to RCP seal LOCA, which cannot be mitigated.

Scenario 4501-2B

This scenario involves a large turbine hydrogen fire with failure of suppression. Collateral damage in the turbine building fails all offsite power from the main switchyard to NB01, NB02, PA01 and PA02. Offsite power is available to NB01 and NB02 from AEPS. The dominant pathway to core damage is through a loss of AFW due to a loss of condensate. Included in this fire is a spurious opening of ADLV0079BB and ADLV0079BA, which drains the CST to minimum tech spec level. After a period of time the CST is empty and non-fire related failures of ESW pumps and room coolers lead to a loss of AFW pump suction. Feed and bleed is similarly failed by the same ESW failures which failed ESW water to the AFW system.

Scenario C10-8s

This scenario is started by a fire in a motor control center, which causes significant cable damage in C-10. Significant train related safety systems are lost by the fire. Offsite power to PA01 and PA02 are also failed by the fire. Opposite train safety systems are unaffected. Offsite power is available to NB01. Core damage is caused by random failures of Train A safety equipment.

Scenario C10-17

This scenario is started by a fire in an electrical panel, which causes significant cable damage in C-10. All Significant train related safety systems are lost by the fire. Offsite power to PA01 and PA02 are also failed by the fire. Opposite train safety systems are unaffected. Offsite power is available to NB01. Core damage is caused by random failures of Train A safety equipment.

Scenario C9-12

This scenario is started by a fire in an electrical panel, which causes significant cable damage in C-9. Significant train related safety systems are lost by the fire. Offsite power to PA01 and PA02 are also failed by the fire. Opposite train safety systems are unaffected. Offsite power is available to NB02. Core damage is caused by random failures of Train B safety equipment.

Scenario RL015/016e Main Control Board RL15/16 Fire with MCR Evacuation

This scenario is a large fire in control board panels RL015 and RL016 in the main control room. Fire is suppressed before is extends beyond the panel RL015/016, but all equipment controlled from this panel is unavailable. Safe shutdown is provided by safety train B equipment from the Auxiliary Shutdown Panel. Offsite power is available to NB02 from AEPS through PB05 and NB0214. Failure to provide safe shutdown from the ASP is attributed to human error and random failures of train B equipment.

Scenario 3801T3

This scenario represents a transient fire in a cable spreading room [C-22], which causes limited damage, but creates some high frequency undesired events. Seal injection isolation valves from all four RCPs [BBHV8141 and BBHV8351] are damaged in this fire. Loss of seal cooling is virtually guaranteed. Spurious opening of ADLV0079BA/BB causes CST drain down to the minimum tech spec water level; ESW makeup is eventually required. Feed and bleed cooling is unavailable due to fire damage to the PORVs.

Scenario 4501-3

This scenario is a catastrophic turbine generator fire which fails all equipment and cables in the Turbine Building, including normal offsite power and offsite power from the AEPS. Automatic fire suppression fails. Random failures of NE01 and NE02 lead to station blackout with no potential credited recovery.

Scenario 3501T11

This scenario represents a transient fire in a cable spreading room [C-21], which causes loss of offsite power to PA01 and PA02 and loss of significant train related safety equipment. AFW is available from PAL02 and PAL01B. Random failures of Train B ESW/CCW and charging system failure to provide seal cooling leads to RCP seal LOCA.

Scenario 3801T2

This scenario represents a transient fire in a cable spreading room [C-22], which causes limited damage, but creates some high frequency undesired events. Seal injection isolation valves from all four RCP's [BBHV8141 and BBHV8351] are damaged in this fire. Loss of seal cooling is virtually guaranteed. Spurious opening of ADLV0079BA/BB causes CST drain down to the minimum tech spec water level; ESW makeup is eventually required. Feed and bleed cooling is unavailable due to fire damage to the PORVs.

Scenario 4501-1B

This scenario involves a large turbine lube oil system fire. Collateral damage in the turbine building fails all offsite power from the main switchyard to NB01, NB02, PA01 and PA02. Offsite power is available to NB01 and NB02 from the AEPS. The dominant pathway to core damage is through a loss of AFW due to a loss of condensate. Included in this fire is a spurious opening of ADLV0079BB and ADLV0079AA, which drains the CST to minimum tech spec level. After some period of time the CST is empty and non-fire related failures of ESW pumps and room coolers lead to a loss of AFW pump suction. Feed and bleed is similarly failed by the same ESW failures which failed ESW water to the AFW system.

Scenario A29-WR

The scenario is caused by a large fire in A-29. Fire modeling was not employed in this room. This scenario fails steam line pressure instrumentation on all steam lines causing spurious opening of SG-ASD's. Fire also fails auxiliary feedwater flow indication on several SG's. Failure of the operator to respond to the loss of instrumentation leads to loss of SG cooling and failure of feed and bleed.

Scenario 4203-0

This is a large floor area transient fire in zone 4203. The normal charging pump and the 3 non-essential service water pumps are failed by this fire. There is no fire induced failure of any safety related equipment or offsite power. Fire risk is driven by random and common cause failures of the essential service water system pumps.

Scenario A-13 WR

This scenario represents large fire in A-13. Fire induced failures include atmospheric steam dumps in Steam generators A, B, D. Fire risk is dominated by fire induced damage to the SG isolation system, with random failure of the intact AFW pumps and failure of feed and bleed cooling.

Plant Modifications related to NFPA 805

The following modifications were identified by the NFPA 805 project. The NFPA 805 project did not consider any other modifications to be necessary.

Modification 07-0066 – the buried carbon steel ESW piping was replaced with high density polyethylene (HDPE) piping. During the piping replacement the cabling associated with DFTE0067A and 68A was relocated to restore the required 20 foot separation criteria. This modification is complete. This modification was not assigned a specific SAMA since it is already complete.

Modification 10-0032 – Installed a non-safety related AFW pump as diverse AFW backup supply to the safety related motor driven and turbine driven pumps. This modification is complete. This modification is related to SAMAs 68 and 78.

Modification 10-0038 – Install four non-safety related diesel generators (8MW) at the electric cooperative substation. Either the electrical cooperative substation or the 4 non-safety diesel generators will be able to power either Safety Related bus in the event of a loss of AC power and failure of the Emergency Diesel Generators. This modification is complete. This modification is related to SAMAs 9 and 14.

Modification 05-3029 – Install lower amperage fuses for various 14 AWG control circuits in the MCR. The majority of the modification centers around the trip circuit fuses on NB, NG, PA, PB, and PG system breakers. This modification is not complete. This modification was added as SAMA 180.

Modification 07-0151 – Install redundant fuses and isolation switches for MCR evacuation procedure OTO-ZZ-00001. This modification is not complete. This modification was added as SAMA 181.

Modification 09-0025 – To protect against multiple spurious operation scenarios, cable runs will be changed to run a single wire in a protected metal jacket such that spurious valve opening due to a hot short affecting the valve control circuit is eliminated for the fire area. This modification will be implemented in multiple fire areas. This modification is not complete. This modification was added as SAMA 182.

Modification 12-0009 – Quick response sprinkler heads in cable chases A-11, C-30, and C-31 will be modified to be in accordance with the applicable requirements of NFPA 13-1976 edition. This modification is not complete. This modification was added as SAMA 183.

b. Section F.3.1.2.3 states the internal events PRA does not include an internal flooding modeling. However, Section F.3.1.1.2 indicates that internal flooding was included in the IPE and in a PRA update as recently as 2004. Discuss the results of the latest applicable internal flooding analysis, the differences from the IPE analysis cited for the internal flooding frequency identified in Section F.3.1.2.3 and potential internal flooding SAMAs based on the latest most applicable internal event flooding analysis.

Response:

The statements referred to in Section F.3.1.2.3 pertain to the fact that, through PRA Update 4B, i.e., the internal events PRA version used in the SAMA analysis, the Callaway internal flooding analysis was not developed into a PRA model that could generate cutsets and thus not integrated into the internal events PRA model. As a result, when the internal events model was perturbed to evaluate SAMAs, the CDF results did not include a contribution from internal flooding; and the stand-alone flooding analysis format did not support any practical method for assessment of SAMAs. Thus, the external events multiplier, developed in Section F.3.1.2.4, includes the PRA Update 4B internal flood numerical CDF of 9.14E-6 per year.

The latest Callaway internal flooding analysis was performed as part of the PRA Update 5 scope of work. The purpose of updating the internal flooding analysis in PRA Update 5 was to move the analysis to Capability Category II of the PRA Standard. Changes made to the analysis method were not dramatic in nature, and included items such as use of a more-recent flood initiator frequency database, more-accurate treatment of non-watertight door flood retention capabilities, and better treatment of operator actions to terminate floods.

As noted in the tabular response to RAI question 1.d.ii, the PRA Update 5 internal flooding analysis CDF is 6.21E-6, or only about two-thirds of the PRA Update 4B internal flooding CDF. Given the approach used to incorporate the internal flooding contribution to CDF in the SAMA analysis, i.e., by inclusion of the internal flooding CDF in the external events multiplier, use of the Update 4B internal flooding CDF in the SAMA analysis is conservative.

SAMA 160 is the only plant specific internal flooding related SAMA. The 1999 internal flooding analysis used as a basis for the SAMA identified only one flood that was below the screening value used. After implementation of the internal flooding task force recommendations, this flood was considered an acceptable risk and no further actions were needed.

Although the Callaway Internal Flooding analysis was updated following the submittal of the License Renewal LAR, there were no significant changes in methodologies used to develop flood scenarios, thus the insights used to assess SAMAs should not be significantly impacted and the existing method for addressing internal flooding in the SAMA analysis remains conservative.

c. Section F.5.2 states that potential enhancements identified in the IPE were included in Table F.5-1. Only four of the five enhancements identified in IPE Section 6.2.1, "Plant Improvements to be Implemented," are included in Table F.5-1 and none of the five enhancements in Section 6.2.2, "Plant Improvements to be Considered," were included. Provide the status and an evaluation of:

i. The missing improvements from IPE Section 6.2.1, addition of procedural guidance and the required hardware to enable the operators to feed one or more steam generators with a diesel driven firewater pump; and,

ii. The five improvements listed in IPE Section 6.2.2.

Response:

The items have been added to the SAMA list as SAMAs 172, 173, 174, 175, 176, and 177 which are included in Attachment 1 to this enclosure. All items have been implemented and therefore screen as Intent Met in the Phase I analysis.

d. Note 1 to Table F.3-2 states, "The current plant procedures and training meet current industry standards. There are no additional specific procedure improvements that could be identified that would affect the result of the human error probability (HEP) calculations. Therefore, no SAMA items were added to the plant specific list of SAMAs as a result of the human actions on the list of basic events with RRW greater than 1.005." This appears to imply that meeting current industry standards is sufficient to indicate that no additional SAMAs are needed.

i. Provide additional information to justify the conclusion stated as indicated above.

ii. Explain the process used to make the determination that there are no opportunities to improve procedures and training. Include in the explanation how human error probability factors were considered (e.g., cognition, resources, timing, and stress level).

iii. Discuss whether any of the risk significant operator action failures could be addressed by options other than training or procedures such as automated functions, testing, and maintenance to reduce failure or event rates, or enhanced documentation. Specifically discuss the potential for automating the function associated with basic event OP-XHE-FO-CCWRHX (OPERATOR FAILS TO INITIATE CCW FLOW TO THE RHR HXS) identified in Table 3-2.

Response:

In order to perform a cost/benefit analysis of any change, the impact on the calculated Human Error Probability must be determined. Discussion with the HRA analysts indicate that based on the current structure and format of the existing procedures, any incremental improvements or changes made to training or procedures would not result in the ability to take additional credit in the HRA because in general full credit is already taken.

Improvements may be possible through re-ordering steps in the EOP network to improve timing, however, since Callaway uses the standardized EOP network significant changes in EOP structure would result in compliance issues with EOP configuration control. The current standardized EOP structure is based on the deterministic safety analysis, not a PRA analysis, thus while there may be PRA improvements there are significant analysis and infrastructure changes that would have to be implemented Industry-wide to change the standardized structure. Any enhancements that could be made within the standardized structure have either already been made at Callaway or would not result in additional significant credit in the HEP determination. With no significant change to the HEP, the benefit of making the change would be negligible.

The note was not intended to state that no opportunities for improvement exist, rather that there would be no calculated dollar value benefit from any improvements made. Plant personnel are always encouraged to use the corrective action process to identify potential improvements. In addition, the PRA group reviews and actively participates in changes made to Operating Procedures to evaluate impact on the PRA as well as suggest improvements.

In general, operator actions credited in the Level 1 PRA, are proceduralized in the EOP and OTO procedure network. The EOP/OTO procedures address both cognition and execution as follows:

- Cognition - specifically they identify the primary cue (instrumentation or alarm needed to make the diagnosis)
- Execution - specifically they identify the tasks needed to accomplish the required action)

The EOP/OTO procedures are highly trained on both in the classroom and to the extent possible in the simulator or through job performance measures. All EOPs are required to be trained on at least once every six years. In general, most EOPS are trained on several times a year in both the simulator and class room training. There is a six week training cycle and each crew will spend one week in simulator and/or class room training during each six week cycle. The trainers review the procedures regularly to identify areas where the training crews have encountered difficulty and update the procedures accordingly. EOP/OTO Writer's manual APA-ZZ-00102 is the guidance document the procedure writers follow to ensure that the procedures are written to be consistent with industry standards.

As part of the HRA task the EOP and OTO procedures are reviewed to ensure that credited operator actions in the PRA are proceduralized in the same context as the EOPs/OTOs. The HRA task accounts for the following:

- Procedure Context -Does the procedure match the modeled PRA scenario,
- Procedure Structure - Response not obtained column format vs paragraphs of instructions,
- Procedure Wording - Does the procedure wording have a double negative,
- Distinction of important steps (boxed, bulleted, bolded, etc),
- Time to reach the required procedure step.

If the HEP is dominated by a single failure mechanism such as an ambiguously worded statement or not enough time to reach the required procedure step, then these findings are passed back to the Callaway training department and procedure revisions are made within the limitations of standardized procedures, as applicable.

The Callaway training department maintains a listing of time critical deterministic and PRA risk significant actions in procedure APA-ZZ-00395. On a defined cycle, the deterministic operator actions are evaluated/validated in the simulator, including timing of events. There is considerable overlap in the deterministic operator actions and the PRA risk significant actions and timing information from these validations is used to evaluate the assumptions in the HRA. This training identifies procedure ambiguities associated with the procedure guidance for most actions credited in the PRA. These completions times are not requirements but are intended to be nominal average estimates that most crews can achieve. Following the completion of a major PRA update APA-ZZ-00395 is updated.

As part of PRA Update 5, all Level 1 post-initiator operator actions were reviewed and updated to align with the current EOP/OTO procedure revisions and training. As part of this update, all risk-significant scenarios were talked through with Callaway trainers and insights from recent simulator training were incorporated into the updated HRA.

The process followed for this HRA update was:

1. Identify – Each PRA scenario was reviewed in the context of the appropriate EOP to ensure that the “as-operated plant” is reflected in the HRA.
2. Define – As part of the definition a feasibility check was performed for each HFE. This included defining both a cognitive and execution procedure, identify the frequency and level of training, showing there is enough time to complete the

action, and there are enough people available to perform all tasks associated with the initiating event.

3. Quantify – The HEP is quantified using the EPRI HRA approach which accounts for a combination of cognitive and execution performance shaping factors.
4. Uncertainty – The uncertainty is addressed both qualitatively and quantitatively in the HRA.

As part of the HRA update no procedure updates to improve the SAMA were identified.

A case was quantified to determine the benefit of automating the initiation of CCW flow to the RHR heat exchangers. This case was evaluated by setting the value of basic event OP-XHE-FO-CCWRHX to 0.0. The benefit of this modification was determined to be $62K with the 95% CDF benefit being $132K. This modification was judged to be not cost-beneficial.

The cost of adding hardware systems to automatically perform the actions represented by important human actions is high. This cost has been shown in a number of SAMA submittals to sometimes be order(s) of magnitude higher than the benefit achieved.

Other non-procedural changes such as additional maintenance and testing would not necessarily reduce risk significant human errors. Most equipment related failures are induced by human errors during testing or maintenance. The benefits of increasing the occurrence of tests and maintenance diminish at the point where additional maintenance or restoration errors are introduced or at the point where undue wear and tear occurs. Callaway's maintenance and testing program uses vendor recommended test and maintenance intervals as well as operating experience in an attempt to optimize mechanical reliability. Randomly increasing test and maintenance over the recommended intervals is perceived to have no mechanical reliability benefit; but would pose an increase in maintenance and restoration errors as well as wear and tear.

e. In Tables 3-2, 3-6, and 3-7, the SAMAs associated with the various basic events in many cases are identified by generic titles such as "Service Water SAMAs," or "Safety Injection SAMAs," rather than citing specific SAMAs that address the failure associated with the basic event. Also, these SAMA categories do not correlate to SAMA categories identified in Table 5-1. For example the categories "Service Water SAMAs," and "Safety Injection SAMAs," are not identified in the fourth column of Table 5-1. In light of this and the fact that only three SAMAs are identified in Table F.5-1 as a result of the importance analysis, the extent of the effort made to identify Callaway specific SAMAs for the important failures is not clear. Within Tables 3-2, 3-6, and 3-7, clarify which SAMA(s) address each specific basic event. Also, please provide a general description of this mapping.

Response:

Tables F.3-2, 3-6 and 3-7 have been revised to address RAIs 5e and 5f, and are provided in Enclosure 3.

f. In importance analyses Tables F.3-2, 3-6, and 3-7, some basic events are not assigned a candidate SAMA but rather with the notation that they are initiating events (i.e., IE-T3, IE-TMSO, IE-S3, IE-T2). Identify SAMAs for these initiating events that either reduce their frequency or mitigate their impact.

Response:

Tables F.3-2, 3-6 and 3-7 have been revised to address RAIs 5e and 5f, and are provided in Enclosure 3.

g. Table F.6-1 indicates that SAMA 3 (add additional battery charger or portable diesel-driven battery charger to existing direct current (DC) system) is screened out on the basis that the intent of this SAMA is met by having two spare battery chargers. This SAMA also includes a diesel driven battery charger. Clarify whether Callaway has a diesel charger that could be considered as part of candidate SAMA and evaluate if appropriate.

Response:

Callaway Energy Center maintains a portable diesel generator capable of supplying any single train of the existing 125V DC (system NK) electrical busses. Emergency Coordinator Supplemental Guidelines, Attachment N, "Temporary Power to NK Swing Chargers" provides procedural guidance on uses the portable generator to provide power to the NK system swing battery chargers.

h. Provide additional information describing the basis for the screening of SAMA 16 (improve uninterruptible power supplies) in Table 6-1. Include explanation of what upgrades were made and for any upgrades made, identify which frontline system those uninterruptible power supplies support.

Response:

The modification replaced the existing Class IE 120 VAC inverters to add a standby regulated 120 VAC source and associated automatic transfer switches to form a complete inverter/uninterruptible power supply (UPS). The automatic transfer function will transfer the 120 VAC load to the standby source in the event of inverter or DC system failure without interruption to the power supply.

The UPSs supply power to the following systems:

- Engineered Safety Feature Actuation System (ESFAS)
- ESFAS indication
- Neutron Flux Monitor
- Ex-Core Neutron Flux Monitor
- Solid State Protection System
- Nuclear Instrumentation (NIS) Cabinet
- Control Room fire isolation panel
- BOP instrument racks
- Safety Related instrument racks
- Subcooling Monitor
- Relay racks

i. Clarify whether remote operation of the atmospheric steam dumps (ASDs), cited in the disposition of SAMA 40 in Table 6-1, is possible and could be credited for risk reduction using the PRA model used to perform the SAMA analysis.

Response:

Emergency Coordinator Supplemental Guidelines, Attachment S "Manual Control of ABPHC0002 to Control ABPV0002," and Attachment T "Manual Control of ABPHC0003 to Control ABPV0003," provide direction to operators on local manual operation of the ASDs. Plant personnel are trained on the use of protective equipment and breathing apparatus that may be required in order to access the controllers in adverse environmental conditions. Given this guidance it may be possible to credit this action to reduce risk; however, this capability is not modeled in the PRA used to develop the SAMA.

j. In Table F.6-1, SAMAs 81, 82, and 83 were screened on the basis that the intent of these heating, ventilation, and air conditioning (HVAC) SAMAs was met at Callaway. In light of the fact that just one general HVAC SAMA (i.e., SAMA 80) was evaluated, please provide further justification for screening out these SAMAs.

Response:

SAMAs 81 and 83 suggested installation of high temperature alarms in the diesel generator rooms and switchgear rooms respectively. These SAMAs were screened as being met because high temperature alarms exist at Callaway for these areas.

SAMA 82 deals with staging portable fans as a backup to switchgear room ventilation. This item was screened as intent met since procedures exist at Callaway for opening doors to provide alternate cooling capability should switchgear room ventilation be lost. Analysis has shown that only opening doors is necessary and the use of portable fans is not required.

As part of the Callaway PRA, room heat-up calculations have been performed to determine those areas that require ventilation to prevent equipment failure. For those areas that require ventilation, the SAMA case HVAC was evaluated by removing the HVAC dependency for these areas in the PRA.

k. In Table F.6-1, SAMA 137 (Provide capability to remove power from the bus powering the control rods) has the following Phase I disposition: "Response procedure in place." Confirm that this procedure includes removing power from the bus powering the control rods.

<u>Response:</u>

Procedure FR-S.1, Response to Nuclear Power Generation/ATWS, provides direction to the Operators to remove power from the busses (PG19 and PG20) powering the control rods. This action is performed within the Control Room and is included in the Licensed Operator training program.

l. In Table F.6-1, SAMA candidate 138 (improve inspection of rubber expansion points on main condenser) is screened out as "Not Applicable" with the disposition that, "No risk significant flooding sources identified in the turbine building." Although the current internal events PRA is stated not to include analysis of internal flooding, the Callaway IPE indicates that internal flooding contributed 31 percent to internal events CDF. Clarify whether this flooding source is possible and whether it can be risk significant. If it can, provide an evaluation for this SAMA.

<u>Response:</u>

The condenser expansion joints are a potential flooding source. The original and updated IPE internal flooding assessment did not identify this flooding source as impacting any safety related equipment. The flooding analysis used to develop the SAMA did not identify this flooding source as risk significant.

The Callaway Internal Flooding analysis was updated again following the submittal of the License Renewal LAR. This updated analysis estimated the risk associated with all circulating water floods, including large expansion joint floods, in the turbine building to be approximately 5E-08/yr which is less than 1 percent of the total flooding CDF.

m. In Table F.6-1, SAMA 141 (provide additional restraints for carbon dioxide (C02) tanks) is combined with other seismic SAMAs (i.e., 154, 155,156,157,158, and 159). None of these SAMAs address this specific issue. Justify why these SAMAs are combined or evaluate them separately.

Response:

The intent was to address generic seismic SAMAs with individual plant specific seismic issues. SAMA 141 could have been screened as "Not Applicable" for Callaway. The topic of SAMA 141 (provide additional restraints for carbon dioxide (CO_2) tanks) was not identified as a seismic issue to be corrected during the IPEEE seismic analysis and Callaway has no CO_2 fire suppression systems within the powerblock. The AEPS diesel generators do have a CO_2 suppression system that is not seismically qualified; however the AEPS diesel generators themselves are not seismically qualified and may not survive a seismic event.

n. In Table F.6-1, SAMA candidate 144 (install additional transfer and isolation switches) for reducing the potential for spurious actuation during a fire is screened out as "Intent Met" based on modification commitments made in the NFPA 805 LAR submittal. NFPA 805 LAR Attachment S does identify such an item (i.e., Item 070151 -Install redundant fuses and switches to prevent multiple spurious actions from stopping or starting safety equipment). However, this modification is specific to selected cables in the Main Control Room to Train B fed from NB02. Justify or evaluate other modifications that would reduce spurious actuations during a fire.

Response:

The following modifications were identified by the NFPA 805 project. The NFPA 805 project did not consider any other modifications to be necessary.

Modification 07-0066 – The buried carbon steel ESW piping was replaced with high density polyethylene (HDPE) piping. During the piping replacement the cabling associated with DFTE0067A and 68A was relocated to restore the required 20 foot separation criteria. This modification is complete. This modification was not assigned a specific SAMA since it is already complete.

Modification 10-0032 – Installed a non-safety related AFW pump as diverse AFW backup supply to the safety related motor driven and turbine driven pumps. This modification is complete. This modification is related to SAMAs 68 and 78.

Modification 10-0038 – Install four non-safety related diesel generators (8MW) at the electric cooperative substation. Either the electrical cooperative substation or the 4 non-safety diesel generators will be able to power either Safety Related bus in the event of a loss of AC power and failure of the Emergency Diesel Generators. This modification is complete. This modification is related to SAMAs 9 and 14.

Modification 05-3029 – Install lower amperage fuses for various 14 AWG control circuits in the MCR. The majority of the modification centers around the trip circuit fuses on NB, NG, PA, PB, and PG system breakers. This modification is not complete. This modification was added as SAMA 180.

Modification 07-0151 – Install redundant fuses and isolation switches for MCR evacuation procedure OTO-ZZ-00001. This modification is not complete. This modification was added as SAMA 181.

Modification 09-0025 – To protect against multiple spurious operation scenarios, cable runs will be changed to run a single wire in a protected metal jacket such that spurious valve opening due to a hot short affecting the valve control circuit is eliminated for the fire area. This modification will be implemented in multiple fire areas. This modification is not complete. This modification was added as SAMA 182.

Modification 12-0009 – Quick response sprinkler heads in cable chases A-11, C-30, and C-31 will be modified to be in accordance with the applicable requirements of NFPA 13-1976 edition. This modification is not complete. This modification was added as SAMA 183.

o. Table 5-1 includes in Note 1 SAMA identification sources to include "D. Expert panel convened to review SAMA analysis." Section 5.5 of the LRA states that "The Callaway plant staff provided plant specific items that were included in the evaluation." Describe this activity in more detail. identifying the individuals involved and how the review was conducted. In addition, please clarify whether the "Expert Panel" was a formal panel or several individuals reviewing material individually.

Response:

The Expert Panel convened to review the SAMA analysis was a formal panel that met over a two day period. The panel was made up of the following personnel:

- Supervising Engineer, Plant Life Extension (Facilitator)
- License Renewal Project Manager
- License Renewal Environmental Lead
- License Renewal Electrical Lead
- PRA Supervisor
- Mechanical Design Engineer
- Operations Supervisor
- Engineering Fix-It-Now Mechanical Engineer
- (Scientech), PRA Engineer/Senior Scientist

The panel first reviewed the Phase I screening results for concurrence. The Panel then reviewed the Phase II screening. In Phase II, each SAMA item was discussed and any questions on how the calculated benefit was determined. In many cases the SAMA analysts had determined an estimated cost based on previous SAMA submittals. If so, the Expert Panel would decide to agree or revise the cost estimate. For other SAMAs, the panel members discussed how the potential modification could be implemented and determined simple minimum cost based on the costs to install similar modifications in the past. The Panel used the 95% CDF sensitivity benefit as the benefit for cost/benefit comparison. The cost estimates and design inputs were only discussed in enough detail to determine that the implementation costs would be significantly greater than the 95% CDF sensitivity benefit.

During this review process, a few members of the Expert Panel brought up possible SAMA items that were generated by the discussions. These were added to the list of SAMA items.

6. <u>With regard to the Phase II Cost-Benefit Evaluations:</u>

a. Provide the percent reduction in off-site economic cost risk (OECR) for each SAMA evaluated in Table F.7-1 and any other SAMAs evaluated in response to RAIs.

Response:

Table F.7-1 has been revised to include the percent reduction in off-site economic cost risk (OECR) for each SAMA item in the table. The updated table is included in Attachment 1 to this enclosure. The percent reduction was calculated by dividing the Total Offsite Benefit for each SAMA case by the Total Offsite Risk calculated for the baseline PRA case.

b. ER Section F.7.2 indicates that an expert panel developed the implementation cost estimates for each of the SAMAs. Describe the level of detail used to develop the cost estimates (i.e., the general cost categories considered). Also, clarify whether the cost estimates accounted for inflation, contingency costs associated with unforeseen implementation obstacles, replacement power during extended outages required to implement the modifications, and maintenance and surveillance costs during plant operation.

Response:

The general categories of costs considered were materials, analyses to support implementation and feasibility, procedure development, replacement power costs, and the costs of ongoing training and surveillance. Inputs such as cost of implementation at other plants and implementation of similar modifications and equipment replacements were also considered. Some estimates included costs of a structure to house the equipment if the Expert Panel felt that sufficient space did not exists within the current plant structures. In general, the discussion for an individual item would start out relatively low and more detail and refinement would take place after comparison of the cost estimate to the benefit at 95% CDF, which was always the highest benefit from the sensitivity evaluations.

The cost estimates did not consider inflation. Contingency costs were not specifically considered. Members used the costs of similar modifications as a basis for their input to the cost estimates and those modifications may have included varying amounts of contingency costs.

c. Confirm which CDF value and contributors (e.g., internal and external) were used to calculate the risk reduction values presented in Table F.7-1. Table F.7-1 presents the reduction in CDF for SAMA 2 as 12.17 percent. This is evaluated as eliminating SBO events. Table F.3-1 presents a value for SBO that is 28 percent of the total. Please explain this discrepancy.

Response:

The values shown in the original Table F.3-1 were incorrect. The original value shown in this table classified sequences with failure of the EDGs with power successfully provided from the AEPS diesel generators as a station blackout. A revised table is provided below that classifies those sequences as Loss of Offsite Power sequences.

Table 3-1. Contributions to Internal Events CDF

Initiating Event Type	Contribution to Internal CDF (/year)
Small LOCA	5.93E-06
Loss of Offsite Power	3.98E-06
SGTR	2.35E-06
RCP Seal LOCA	8.63E-07
Reactor Trip	7.88E-07
Station Blackout	7.85E-07
Intermediate LOCA	3.67E-07
All Steam Line Breaks	3.35E-07
Anticipated Transient without Scram (ATWS)	2.04E-07
ISLOCA	1.73E-07
Loss of Feedwater	1.65E-07
Very Small LOCA	1.29E-07
Loss of CCW	1.20E-07
Loss of SW	1.15E-07
Feedwater Line Break	9.01E-08
Loss of DC Vital Bus	6.93E-08
PORV Fails to Reclose	4.52E-08
Large LOCA	4.21E-08
Total	1.66E-05
LOCA = loss of coolant accident; SGTR = steam generator tube rupture; RCP = reactor coolant pump; CCW = component cooling water; SW = service water; DC = direct current; PORV = power operated relief valve	

d. Clarify modeling assumptions used for SAMA cases in which failures are eliminated (e.g., service water pumps) by indicating which failures were eliminated including whether this includes support failure (e.g., mechanical failure of service water pumps and support failures such as alternating current (AC) power supply to service water pumps).

Response:

Descriptions of the changes made to the PRA model for each SAMA case are shown below. Preparation of the response determined that cases NOSGTR and NOSLOCA needed revision and these cases were re-quantified. New SAMA cases created in response to the RAIs have been added. The revised results are included in the revised tables F7-1, F8-1, and F11-1 which are included in Attachment 1 to this enclosure.

NOATWS

SB-ICC-AF-RXTRIP
BB-BKR-CC-TRPBKR
BB-RCA-WW-RCCAS

The above basic events represent reactor trip failure modes and are set to 0.0 in BED file SAMANOATWS.BED. There are no modeled support systems associated with these failure events.

This case is used to determine the benefit of eliminating all Anticipated Transient Without Scram (ATWS) events. For the purposes of the analysis, a single bounding analysis was performed which assumed that ATWS events do not occur.

NOSGTR

IE-TSG
L2-SGT-VF-PISGR
L2=SGT-VF-TISGR

The above basic events were set to 0.0. The basic events represent the Steam Generator Tube Rupture initiating event and the probability of pressure and thermally induced tube ruptures. This allows evaluation of various possible improvements that could reduce the risk associated with SGTR events. For the purposes of this analysis, a single bounding analysis was performed which assumed that SGTR events do not occur.

INSTAIR

KA-PSF-VF-ISTAIR
KA-PSF-VF-TKA02
KA-PSF-VF-TKA03
KA-PSF-VF-TKA04
KA-PSF-VF-TKA05

The above basic events are set to 0.0 in BED file SAMAINSTAIR.BED. The PRA model does not contain detailed modeling of the instrument air system. The first basic event represents failure of the instrument air system. The others represent failures of the steam generator PORV backup nitrogen supply. Support system failures were not directly considered in this case.

This case is used to determine the benefit of replacing the air compressors. For the purposes of the analysis, a single bounding condition was performed, which assumed the station air systems do not fail.

NOLOSP

IE-T1

The above basic event is set to 0.0 in BED file SAMANOLOSP.BED. This eliminates the Loss of Offsite Power initiating event.

This case is used to determine the benefit of eliminating all Loss of Offsite Power (LOSP) events, both as the initiating event and subsequent to a different initiating event. This allows evaluation of various possible improvements that could reduce the risk associated with LOSP events. For the purposes of the analysis, a single bounding analysis was performed which assumed that LOSP events do not occur.

CCW01

EG-MDP-CR-EGPMP3
EG-MDP-DR-EGPMP4
EG-MDP-DS-EGPMP3
EG-MDP-DS-EGPMP4
EG-MDP-FR-PUMPA
EG-MDP-FR-PUMPB
EG-MDP-FR-PUMPC
EG-MDP-FR-PUMPD
EG-MDP-FS-PUMPA
EG-MDP-FS-PUMPB
EG-MDP-FS-PUMPC
EG-MDP-FS-PUMPD

The above basic events are set to 0.0 in BED file SAMACCW01.BED. These basic events represent failures of the CCW pumps. This case did not consider support system failures.

This case is used to determine the benefit of improvement to the CCW system by assuming that CCW pumps do not fail.

FW01

IE-T2

The above basic event is set to 0.0 in BED file SAMAFW01.BED. This eliminates the loss of feedwater initiating event.

Eliminate loss of feedwater initiating events. This case is used to determine the benefit of improvements to the feedwater and feedwater control systems.

NOSLB

IE-TMSI
IE-TMSO

The above basic events are set to 0.0 in BED file SAMANOSLB.BED. This eliminates the steam line break initiating events, both inside and outside containment.

This case is used to determine the benefit of installing secondary side guard pipes to the Main Steam Isolation Valves (MSIVs). This would prevent secondary side depressurization should a Steam Line Break (SLB) occur upstream of the MSIVs. For the purposes of the analysis, a single bounding analysis was performed which assumed that no SLB events occur.

CHG01

The fault trees were modified to remove the cooling water dependency for the charging pumps. This assumes the charging pumps are not dependent on cooling water. This case is used to determine the benefit of removing the charging pumps dependency on cooling water. No other support system modifications were made.

SW01

The Essential Service Water system fault trees were modified to remove the dependency on DC power. This assumes the Essential Service Water pumps are not dependent on DC power. This case is used to determine the benefit of enhancing the DC control power to the Essential Service Water pumps. No other support system modifications were made.

NOSBO

NE-DGN-DR-NE01-2
NE-DGN-DS-NE01-2
NE-DGN-FR-NE01
NE-DGN-FR-NE01-2
NE-DGN-FR-NE01-8
NE-DGN-FR-NE0110
NE-DGN-FR-NE0112
NE-DGN-FR-NE02
NE-DGN-FR-NE02-2
NE-DGN-FR-NE02-8
NE-DGN-FR-NE0210
NE-DGN-FR-NE0212
NE-DGN-FS-NE01
NE-DGN-FS-NE02
NE-DGN-TM-NE01
NE-DGN-TM-NE02

The above basic events represent failures of the emergency diesel generators and are set to 0.0 for this case. Failures of the EDG support systems were not modified. This case is used to determine the benefit of eliminating all Station Blackout (SBO) events. This allows evaluation of possible improvements related to SBO sequences. For the purpose of the analysis, a single bounding analysis is performed that assumes the emergency AC power supplies do not fail.

LOCA05

IE-A
IE-S1
IE-S2
IE-S3

The above basic events represent the initiating events Large LOCA, Medium LOCA, Small LOCA, and Small-Small LOCA. For this case these events were set to 0.0 to represent that piping system LOCAs do not occur. This case is used to determine the benefit of eliminating all LOCA events related to piping failure (no change to non-piping failure is considered).

NOSLOCA

IE-S2
IE-S3
BB-PRV-OC-V455A
BB-PRV-OC-V456A

The above basic events were set to 0.0. These basic events represent the small and small-small LOCA initiating events and the failure of the PORVs to close following an opening. These values were applied to the model modification used to evaluate RCP seal LOCAs. No other support system modifications were made.

This case is used to evaluate the elimination of all small LOCAs, from pipe breaks, stuck open PORV, and RCP seal LOCA.

H2BURN

L2-CNT-VF-CFE1
L2-CNT-VF-CFE5

The above basic events were set to 0.0. These basic events represent the probability of containment failure due to hydrogen burns.

Assume hydrogen burns and detonations do not occur. This case is used to determine the benefit of eliminating all hydrogen ignition and burns.

RCPLOCA

Created new data file SAMARCPLOCA.BED with a single basic event, NORCPLOCA, with value 0.0. The fault tree for RCP seal LOCA was modified to add this event under the AND gate representing causes of RCP seal failures. This logically eliminates all seal failure causes from the calculation.

This case is used to determine the benefit of eliminating all Reactor Coolant Pump (RCP) seal loss of coolant accident (LOCA) events. This allows evaluation of various possible improvements that could reduce the risk associated with RCP seal LOCA and other small LOCA events.

LOCA02

Created new BED file SAMAEVNTZERO.BED with a single basic event, EVNTZERO, with value 0.0. Modified fault trees 1OF4HPI, 1OF4HPR, and 14HPI1S to add event EVNTZERO as input to top AND gate. Modified fault trees HPCI-1, HPCI-2, HPCR-1, and HPCISBO to change to single top gate with only event EVNTZERO as input. No changes to support system logic were made.

This case is used to determine the benefit of no failures of high pressure injection/recirculation systems. This allows evaluation of various possible improvements that could reduce the risk associated with high pressure injection/recirculation failures.

LOCA12

BG-MDP-DR-CCPS
BG-MSP-DS-CCPS
BG-MDP-FR-CCPA
BG-MDP-FR-CCPB
BG-MDP-FR-NCP
BG-MDP-FS-CCPA
BG-MDP-FS-CCPB
BG-MDP-FS-NCP
BG-MDP-TM-CCPA
BG-MDP-TM-CCPB
EM-MDP-DR-SIPMPS
EM-MSP-DS-SIPMPS
EM-MDP-FR-PEM01A
EM-MDP-FR-PEM01B
EM-MDP-FS-PEM01A
EM-MDP-FS-PEM01B
EM-MDP-TM-PEM01A
EM-MDP-TM-PEM01B

The above basic events represent failures of safety injection and charging pumps and are set to 0.0 for this case. Failures of the support systems were not modified. This case is used to determine the benefit of no failures of high pressure injection/recirculation pumps. This allows evaluation of various possible improvements that could reduce the risk associated with high pressure injection/recirculation pump failures.

CONT02

The results equation for failure of containment isolation was set to 0.0.

This case is used to determine the benefit of no containment isolation failures. This allows evaluation of various possible improvements that could reduce the risk associated with all containment isolation failures.

LOCA04

BN-TNK-FC-RWSTUA

The above basic event represents unavailability of the RWST and is set to 0.0. No support system modifications were made.

This case is used to determine the benefit of additional RWST inventory. This allows evaluation of various possible improvements that could reduce the risk associated with RWST inventory by assuming that the RWST does not run out of water.

CONT01

The results equation for late containment failure due to containment overpressure was set to 0.0.

This case is used to determine the benefit of no containment overpressure failures. This allows evaluation of various possible improvements that could reduce the risk associated with all containment overpressure failures.

LOCA03

EJ-MDP-DR-EJPMPS
EJ-MDP-DS-EJPMPS
EJ-MDP-FR-PEJ01A
EJ-MDP-FR-PEJ01B
EJ-MDP-FS-PEJ01A
EJ-MDP-FS-PEJ01B

The above basic events represent failures of the RHR pumps and are set to 0.0 for this case. Failures of the support systems were not modified.

This case is used to determine the benefit of no failures of low pressure injection/recirculation pumps. This allows evaluation of various possible improvements that could reduce the risk associated with low pressure injection/recirculation pump failures.

SW02

Created new BED file SAMASW02.BED with the following basic events set to 0.0.

EF-MDP-DR-EFPMPS
EF-MDP-DS-EFPMPS
EF-MDP-FR-PEF01A
EF-MDP-FR-PEF01B
EF-MDP-FS-PEF01A
EF-MDP-FS-PEF01B

EF-PSF-TM-ESWTNA
EF-PSF-TM-ESWTNB

The above basic events represent failures of the ESW pumps and are set to 0.0 for this case. Failures of support systems were not modified. This case is used to determine the benefit of no failures of Essential Service Water pumps.

DC01

The AFW fault trees were modified to remove the dependency on DC power for the TDAFW pump. No other support system modifications were made.

This case is used to determine the benefit of removing TDAFW pump dependency on DC power. This allows evaluation of various possible improvements that could reduce the risk associated with the TDAFW pump dependency on DC power.

CCW02

This case combined cases CCW01 and SW02. This represents failures of the CCW and ESW pumps. No support system modifications were made.

Sets all CCW pumps and ESW pumps to 0.0 to evaluate the benefit of backup cooling water supplies.

ISLOCA

The results spreadsheet release category LERF-IS was set to 0.0.

This case is used to determine the benefit of eliminating intra-system LOCA failures. This allows evaluation of various possible improvements that could reduce the risk associated with all intra-system LOCA failures.

LOSP1

Basic event TORNADO-T1-EVENT was set to 0.0. This basic event is the probability that a Loss of Offsite Power Event was caused by a tornado.

This case is used to determine the benefit of no tornado-related failures of the Alternate Emergency Power System (AEPS). This allows evaluation of various possible improvements that could reduce the risk associated with tornado induced failures of AEPS by providing tornado protection for the AEPS diesel generators and associated circuits.

DEPRESS

In Fault tree DEPRESS.LGC gate GDEP100 was changed to an AND gate and a new basic event, EVENTZERO, was added as an input and set to 0.0.

This case is used to determine the benefit of no failures of depressurization. This allows evaluation of various possible improvements that could reduce the risk associated with depressurization failures by eliminating depressurization failures.

LOCA06

Basic event IE-A was set to 0.0. This eliminates the Large LOCA initiating event. No support system changes were made.

This case is used to determine the benefit of eliminating Large LOCAs. This allows evaluation of various possible improvements that could reduce the risk associated with all Large LOCA events.

HVAC

The fault trees for AFW, Safety Injection, RCP Seal Cooling, Emergency Diesel Generators, and DC power were modified to remove the dependency on room cooling. No other support system changes were made.

These changes eliminate various HVAC dependencies. This allows evaluation of various possible improvements that could reduce the risk associated with failures of various HVAC systems.

FB01

The fault trees were changed to replace the logic requiring two PORVs for successful feed and bleed with the logic for only one PORV required for feed & bleed. No support system modifications were made.

This case was used to evaluate modifying the PORVs such that only one PORV is required for Feed and Bleed.

PORV

BB-PRV-CC-V455A
BB-PRV-CC-V456A

The above basic events were set to 0.0. These basic events represent failure of the PORVs to open. No support system changes were made.

This case was used to evaluate improvements that lower the probability of PORVs failing to open.

EDGFUEL

The fault trees for the emergency diesel generators were modified to eliminate the dependency on the fuel oil transfer system. No other support system modifications were made.

This case was used to evaluate the addition of a gravity feed EDG fuel oil tank.

FW02

AE-CKV-CC-AEV120
AE-CKV-CC-AEV121
AE-CKV-CC-AEV122
AE-CKV-CC-AEV123
AE-CKV-DF-V120-3

The above basic events represent failure of the feedwater check valves to open and were set to 0.0 for this case. No support system changes were made.

This case was used to evaluate improvements that lower the probability of feedwater check valves failing to open.

SW03

The fault trees for the SW System Train A were modified to add AEPS as a possible power source. No other support system modifications were made.

This case was used to evaluate adding the ability to power the normal service water pumps from the AEPS.

HVAC02

VD-FAN-DR-GD02AB
VD-FAN-DS-GD02AB
VD-FAN-FR-CGD02A
VD-FAN-FR-CGD02B
VD-FAN-FS-CGD02A
VD-FAN-FS-CGD02B

The listed basic events represent failures of the fans for the UHS cooling tower electrical room and were set to 0.0 for this case. No other support system modifications were made.

This case was used to evaluate adding additional UHS cooling tower electrical room HVAC.

CST01

The probability for basic event AD-TNK-FC-CSTUNA (Condensate Storage Tank Unavailable) to 0.0. Supporting analysis for the PRA shows that the CST does not deplete within the 24 hour PRA mission time so this is the only PRA manipulation required. No other support system modifications were made.

This case was used to evaluate adding a second CST or expanding the capacity of the current CST.

HEP

Basic event OP-XHE-FO-CCWRHX (Operator fails to initiate CCW flow to the RHR HXs) was set to 0.0. No other support system modifications were made.

This case was used to support the response to RAI 5.d.iii.

RAI7a

Basic event L2-SGT-VF-TISGR was set to 0.0 for this evaluation. This basic event represents the probability of a thermally induced steam generator tube rupture. No other support system modifications were made.

This case was used to evaluate the impact of a procedure change to prevent the operators from clearing the RCS cold leg loop seal following core damage (RAI 7.a).

SLIS

Basic events SA-ICC-AF-MSLIS, SA-ICC-AF-MSLIS1, SA-ICC-AF-MSLIS4 were set to 0.0. These basic events represent the failure of the main steam line isolation system. No support system modifications were made.

This case is used to evaluate improvements in the main steam line isolation system.

FWCCW2

The RHR fault trees were modified to add fire water as a backup source of cooling to the RHR heat exchangers. The fire water pumps and system do not appear in the PRA. To simulate the use of the fire water pumps and operator actions to perform the temporary hookup, a single basic event with failure probability of 0.1 was placed in the fault trees.

This case is used to evaluate the benefit of providing a temporary hookup of fire water to the RHR heat exchangers.

e. For certain Phase II SAMAs listed in Table F.7 -1, the information provided does not sufficiently describe the associated modifications to clearly identify what is included to justify the cost estimates. Provide a more detailed description of the modifications and cost estimates for SAMAs 11, 15, 64, 94, 104, 116, 163, and 164.

Response:

The process used to determine the cost of potential modification was to have the Expert Panel draw on their knowledge and experience with previous modifications and costs estimated in SAMA submittals from other plants. The modification needed was discussed in general terms unless the cost appeared to be close to being potentially cost-beneficial. In some cases, preliminary cost estimates existed for proposed design modifications that would implement either the SAMA being described or a similar modification.

For SAMA 11, the cost estimate considered the analysis required to support the implementation, materials to be purchased and pre-staged, the development of procedures to support the implementation of the cross-tie, the initial and continuing training of personnel on how to implement the cross-tie procedure, and the cost of periodic inspections to ensure all pre-staged material and equipment is present.

For SAMA 15, the cost estimate was based on a preliminary design that was considered at the time the AEPS diesel generators were constructed and installed.

The benefit of SAMA 64 would be better estimated by using the benefit calculated for case SW02 (no failures of ESW pumps) rather than case CCW01 (no failures of the CCW pumps). Using this case benefits, the development of temporary procedures to use fire water to cool the CCW heat exchangers is potentially cost beneficial. This SAMA has been added to the list of potentially cost-beneficial SAMAs.

The discussion for SAMA 94 determined that a suitable containment penetration that could be used for this filtered vent does not exist. It was estimated that creating a new containment penetration would cost in excess of $1M. The cost of purchase and installing the new equipment was estimated to cost in excess of $1M. No further costs were added to estimate, so the additional costs of procedures, training, on-going testing and inspection were not included.

The discussion for SAMA 104 noted that most of the piping to be inspected is located inside containment. In order to reduce the radiation dose during the inspections, the plant would be required to either significantly reduce power or shut down. The replacement power costs required to perform the inspections was estimated to be in excess of $2M.

The discussion of SAMA 116 noted that the current plant design and licensing basis require the floor drains in these rooms to be open. The cost of analysis and license changes required to support implementation of this SAMA item were determined to be in excess of $1M.

The discussion for SAMA 163 noted that the feedwater check valves were previously replaced to improve reliability. The cost of replacement in the past exceeded $500K.

The cost estimate for SAMA 164 included an estimate of the cost of cabling and wall penetrations for running the cable. Due to the distance involved, the cost was estimated to exceed $500K.

f. For certain Phase II SAMAs listed in Table F.7-1, the calculated benefit does not seem consistent with the percent reduction in CDF or off-site dose or there was no CDF or off-site dose information to compare to the calculated benefit. Provide corrections or more justification for the benefit calculated for SAMAs 39, 160,161, 162, 163, 164, and 171.

Response:

The information was added to table F.7-1 for SAMAs 161, 162, 163, 164, and 171.

The original submittal included this information for SAMA 39, but did not for SAMA 29. This response assumes that the question is concerning SAMA 29.

The Expert Panel concluded that SAMA items 29 and 160 were potentially cost-beneficial without determining an actual cost or benefit. These two items were considered to be relatively low cost for implementation and should therefore be entered into the Callaway long-range plan development process for further consideration. Determination of a numerical benefit of SAMA item 160 is not directly possible since the existing Callaway Internal Flooding study is not quantifiable and does not include this flood propagation path.

g. In Table 7-1, SAMA 1 (add additional DC battery capacity) is evaluated by eliminating turbine driven auxiliary feed water (TDAFW) pump dependency on DC power while SAMA 2 (replace lead-acid batteries with fuel cells) is evaluated by eliminating all SBO. For SAMA 1 and SAMA 5 (provide DC bus cross ties also evaluated by eliminating the TDAFW pump DC dependency), describe whether the TDAFW pump availability is the only impact of the loss of DC. Both SAMAs 1 and 2 extend DC power availability during SBO. Explain the reasons for the different evaluations that do the same thing.

Response:

Tables F7-1 and F8-1 have been revised, and included in Attachment 1, to reflect case NOSBO for SAMA 1. The Expert Panel cost estimate considers the cost of material (batteries, chargers, and cables), a structure to house the new equipment, and ongoing battery monitoring and testing. For SAMAs 1, 2, and 5 in addition to the TDAFW pump dependency, loss of DC impacts the availability of instrumentation. Emergency Coordinator Supplemental Guidelines exist for the use of portable generators to provide backup power on extended SBO events. This backup portable power is not credited in the PRA.

h. SAMA 15 (install tornado protection on gas turbine generator) is evaluated by SAMA case LOSP1 which is described as leading to no tornado LOSP events. Given Callaway has alternate emergency power system (AEPS) diesel generators rather than a gas turbine, clarify the model changes made and their applicability to this SAMA.

Response:

The AEPS diesel generators are not located in a tornado resistant building. The intent of this SAMA was interpreted to be providing a tornado resistant building for the AEPS diesel generators. The case LOSP1 was evaluated by setting basic event TORNADO-T1-EVENT (conditional probability that a Tornado event initiates a LOSP event and directly causes the loss of AEPS) to a probability of 0.0, which eliminates tornado induced AEPS failures. This allows the determination of the benefit of providing tornado protection for the AEPS diesel generators.

i. In Table F.7-1, SAMA 24 (bury off-site power lines) is shown as costing >$3M and as not being cost beneficial. However, the potential benefit of this SAMA is high ($1.2M) and the estimated cost of this SAMA reported in the Seabrook ER (a recent Westinghouse PWR-4 submittal) is lower (>$1 M). Provide a more detailed description of this modification and additional justification for the estimated cost.

Response:

In order to provide the necessary benefit, the offsite power lines would need to be buried the full length of the line to next transmission substation. The nearest transmission substation is approximately 21 miles from the site. The industry accepted cost estimate for burying power lines is approximately $1M per mile, thus the modification would cost approximately $21M.

j. Provide additional information on the changes made for SAMA Case LOCA 12 used to evaluate SAMAs 25, 26, and 39. Describe what modeling change was made to eliminate failures of the charging or SI pumps. Include as part of this description whether these assumed failures are limited to LOCAs or if they include failure due to loss of AC.

Response:

SAMA case LOCA12 was evaluated by creating a new BED file SAMALC12.BED with the following basic events set to 0.0.

BG-MDP-DR-CCPS
BG-MSP-DS-CCPS
BG-MDP-FR-CCPA
BG-MDP-FR-CCPB
BG-MDP-FR-NCP
BG-MDP-FS-CCPA
BG-MDP-FS-CCPB
BG-MDP-FS-NCP
BG-MDP-TM-CCPA
BG-MDP-TM-CCPB
EM-MDP-DR-SIPMPS
EM-MSP-DS-SIPMPS
EM-MDP-FR-PEM01A
EM-MDP-FR-PEM01B
EM-MDP-FS-PEM01A
EM-MDP-FS-PEM01B
EM-MDP-TM-PEM01A
EM-MDP-TM-PEM01B

These basic events represent direct failures of the two SI and three high pressure charging pumps. Setting these basic events to 0.0 eliminates failures of the pumps to start or run as well as unavailability due to testing or maintenance. This case considered eliminating only failures of the actual pumps and did not eliminate failures due to loss a support system such as AC power. Modifying the model in this way eliminates these direct pump failures from all accident sequences which call for the operation of these pumps.

k. Provide additional information on the changes made for SAMA Case LOCA03 used to evaluate SAMA 28. Describe what modeling change was made to eliminate failures of the low pressure pumps. Include as part of this description whether these assumed failures are limited to LOCAs or if they include failure due to loss of AC.

Response:

SAMA case LOCA03 was evaluated by creating a new BED file SAMALOCA03.BED with the following basic events set to 0.0.

EJ-MDP-DR-EJPMPS
EJ-MDP-DS-EJPMPS
EJ-MDP-FR-PEJ01A
EJ-MDP-FR-PEJ01B
EJ-MDP-FS-PEJ01A
EJ-MDP-FS-PEJ01B

These basic events represent direct failures of the low pressure injection pumps. Setting these basic events to 0.0 eliminates failures of the pumps to start or run. This case considered eliminating only failures of the actual pumps and did not eliminate failures due to loss a support system such as AC power. Modifying the model in this way eliminates these direct pump failures from all accident sequences which call for the operation of these pumps.

l. In Table F.7-1 the benefit for SAMA 39 appears to be excessively high (i.e., $748K) when compared to other similar SAMA benefits. Provide corrections as needed.

Response:

The current benefit of $748K shown in Table F7-1 for SAMA 39 is a typographical error. The correct benefit value for SAMA 39 is $48K. Table F7-1 has been corrected and included in Attachment 1.

m. Table F.7-1 indicates that SAMA 46 (add a service water pump) was modeled by assuming there were no failures of essential service water (ESW) pumps. Clarify whether modeling of this SAMA case includes ESW pump unavailability due to test and maintenance.

Response:

The original PRA case SW02 did not include test and maintenance events. The case was modified to include setting the test and maintenance events to 0.0. The results of the revised case are reflected in the revised Tables F7-1, F8-1, and F11-1 which are included in Attachment 1 to this enclosure.

n. In Table F.7-1, SAMA 94 (install a filtered containment vent to remove decay heat) is shown as >$2M and as not being cost beneficial. However, the potential benefit of this SAMA is high ($1.2M) and the estimated cost of this SAMA reported in the Seabrook ER is lower (>$500K). Provide a more detailed description of this modification and additional justification for the estimated cost.

Response:

The Expert Panel discussion for SAMA 94 determined that a suitable containment penetration that could be used for this filtered vent does not exist. It was estimated that designing and licensing of a new containment penetration or modification of an existing penetration would cost in excess of $1M. The cost of procurement, installation and maintenance (including on-going testing and inspections) of the new equipment was estimated to cost in excess of $1M. No further costs were added to estimate, so the additional costs of procedures and training were not included.

o. In Table F.7-1, SAMA 113 (increase leak testing of valves in interfacing systems (IS) LOCA paths) is shown as costing >$1 M and as not being cost beneficial. However, the potential benefit of this SAMA is moderate ($123K) and the cost of this SAMA seems high, as it does not require hardware modification. The Seabrook ER reports an estimated cost of >$100K for this SAMA. Provide a more detailed description of this modification and additional justification for the estimated cost.

Response:

The containment isolation valves in the ISLOCA pathways are currently tested every refueling outage. In order to test these valves the plant must be in Cold Shutdown/Refueling conditions when the valves are accessible and the systems can be aligned/configured to allow installation of test equipment and the performance of the testing. Leak testing on a more frequent basis would require plant shutdown. The cost of replacement power to support shutdowns to test the valves was estimated to be significantly greater than $1M. Callaway currently does not have any regularly scheduled mid-cycle outages. Mid-cycle forced or scheduled outages could produce an opportunity for performing these tests, however any extension of these outages in order to perform leak testing on these valves would quickly accumulate costs that would make this testing not cost-beneficial. Replacement power costs for outage extension is generally estimated to be $1M per day.

p. In Table F.7-1, SAM A 119 (institute a maintenance practice to perform a 100 percent inspection of steam generator tubes during each refueling outage) is shown as costing >$3M and as not being cost beneficial. However, the potential benefit of this SAMA is high ($1.2M) and the cost of this SAMA seems high, as it does not require hardware modification. The Seabrook ER reports an estimated cost of >$500K for this SAMA. Provide a more detailed description of this modification and additional justification for the estimated cost.

Response:

Due to the recent replacement of steam generators and the associated reduced inspection requirements, the expert panel estimated that performing a 100% inspection every refueling outage would extend the duration of many outages. In addition, testing of steam generator tubes requires considerable radiological dose, testing equipment costs and vendor costs for data analysis and reporting. The sum of these costs is in excess of the estimated $3M for this SAMA.

q. Section F.11 states that the RCPLOCA modeling case "allows evaluation of various possible improvements that could reduce the risk associated with RCP seal LOCA and other small LOCA events." As for other SAMA cases, provide a description of the specific modeling assumptions made to determine the percent reduction in CDF and off-site dose.

Response:

SAMA case RCP-LOCA was evaluated by using the following changes to the baseline PRA model:

The fault tree that develops Reactor Coolant Pump (RCP) seal loss of coolant accident (LOCA) cutsets, RCP-1, was modified by adding a new basic event called NORCPLOCA to the top gate of the fault tree. When failed, basic event NORCPLOCA prevents the RCP seal LOCA portion of the model from being solved. A new BED file, SAMARCPLOCA.BED, was created with basic event NORCPLOCA failed by setting it to a value of 0.0. This evaluates elimination of all RCP seal LOCA events that are caused by failure of seal cooling and injection except those which occur as a result of a support system initiating event such as loss of CCW.

Case RCP-LOCA used SAMARCPLOCA.BED and the modified fault tree modeling to determine the benefit of eliminating all RCP seal LOCA accident sequences.

r. Section F.8.2 indicates that the uncertainty factor used for the ratio of the 95th percentile value to the mean value of the CDF is 2.11. In Table F.8-1, the ratio of the base cost benefit to the 95th percentile case for SAMAs 91, 93, and 94 appears to be low (i.e., 1.4). Please explain this apparent discrepancy, or if this is a mistake, recalculate the 95th percentile benefit for these three SAMAs.

Response:

The nominal case benefit listed for SAMAs 91, 93, and 94 in Tables F7-1 and F.8.2 is a typographical error. The correct nominal benefit for each is $793K. The listed 95% benefits of $1.7M are correct. Tables F.7-1 and F.8-1 have been revised, and included in Attachment 1, to show the corrected nominal benefit.

s. Table F.7-1 reports the baseline benefit for SAMA 136 to be $53K, whereas Table F.8-1 reports this value as $63K. Provide corrections as needed.

Response:

The current benefit shown in Table F.7-1 for SAMA 136 is a typographical error. The correct benefit value for this SAMA is $63K. Table F.7-1 has been corrected and included in Attachment 1.

t. Table F.7-1 reports a 95th percentile benefit for SAMA 24 as 2.4M but should be 2.5M. Provide corrections as needed.

Response:

As Table F.7-1 does not list a 95th percentile value, it is assumed that the RAI is referring to Table F.8-1. The current 95th percentile CDF benefit shown in Table F.8-1 for SAMA 24 is a typographical error. The correct 95th percentile CDF benefit value for SAMA 24 is $2.6M. Table F.8-1 has been corrected and included in Attachment 1.

u. The NRC staff has been unable to find a description for SAMA case CST01 identified in Tables F.7-1 and F.8-1 of Section F.11. As for the other SAMA cases, provide a description of the case and the specific modeling assumptions made to determine the percent reduction in CDF and off-site dose.

Response:

Case CST01 changed the probability for basic event AD-TNK-FC-CSTUNA (Condensate Storage Tank Unavailable) to 0.0 representing perfect reliability of the CST. Supporting analysis for the PRA shows that the CST does not deplete within the 24 hour PRA mission time so this is the only PRA manipulation required to emulate additional CST capabilities such that a CST source is always available.

v. The Section F.11 Annex defines SAMA case "HVAC" as eliminating various HVAC dependencies. Identify which HVAC systems this applies to and how the benefit was calculated. Also, confirm which dependencies this applies to.

Response:

This case evaluated equipment dependencies from the following HVAC systems:

- Motor driven AFW pumps
- Charging pumps
- EDGs
- DC switchgear

The benefit was calculated by modifying the PRA model to remove the HVAC dependency logic from the fault trees and solving the PRA model.

Case HVAC02 determined the benefit from eliminating all failures of the Ultimate Heat Sink Cooling Tower electrical room HVAC.

All other equipment was shown to not require room cooling through heat-up calculations performed to support the PRA.

Procedural guidance exists for opening doors to the DC switchgear rooms following loss of HVAC. SAMA 80 has been modified to be potentially cost beneficial to provide procedural guidance to open doors or provide temporary ventilation to the EDGs, motor driven AFW pumps, and charging pumps. Loss of HVAC is only an issue for the EDGs if outside ambient temperature is above 60°F.

7. <u>With regard to Alternative SAMAs</u>

a. A note at the end of Table F.5-1 indicates that recent industry submittals of like-kind plants (i.e., Wolf Creek, South Texas, Diablo Canyon, and Seabrook) were used as a source of candidate SAMAs. The extent to which these submittals were examined is not clear, as only two SAMA candidates were identified in Table F.5-1 as being from these sources (i.e., SAMA 162 and 165). Also, it appears that a cost beneficial SAMA identified in the Diablo Canyon submittal might represent an unevaluated SAMA candidate for Callaway (i.e., SAMA 24 – Prevent clearing of

RCS cold leg water seals). Describe the extent to which the four cited SAMA submittals were used as sources to generate candidate SAMAs and evaluate each SAMA determined to be cost beneficial in those submittals or show how they could be screened out using criteria presented in ER Section F.6.0. If the SAMA review for a submittal has been completed, use the cost beneficial SAMAs as reported in the respective site specific volume of NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants."

Response:

Response to be provided by separate correspondence.

b. SAMA 64 (implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie) is evaluated by eliminating CCW pump failures. Consider a similar SAMA that provides fire water to the ESW system.

Response:

SAMA 64 was revised to evaluate the benefit of a temporary hookup of fire water as backup on loss of CCW cooling to the RHR heat exchangers. This determined the benefit to be $104K with a 95% CDF benefit of $220K. This SAMA is considered potentially cost beneficial and has been added to the list of potentially cost beneficial SAMAs.

SAMA 186 was added to evaluate procedures to provide fire water to the ESW system. This SAMA was considered potentially cost beneficial based on the 95% CDF benefit. Implementation of this SAMA will cost significantly more than a procedure change since it would require replacement of the existing fire pumps with larger pumps.

c. Table 7-1 indicates that elimination of all HVAC dependencies for SAMA 80 results in a 6 percent reduction in CDF. The individual HVAC failures listed in Table 3-2 appear to involve unrelated pieces of equipment in various rooms or buildings. Discuss the possibility of lower cost alternatives that address the more important contributors to CDF. Note that two of the above cited failures (VD-FAN-FR-CGD02A and -CGD02B) appear to be the reason for SAMA Case HVAC02 described on Page F-109. This case is not used in the Phase II analyses described in Table 7-1.

Response:

SAMA 178 was added to evaluate the cited fan failures and is evaluated using SAMA case HVAC02. Creating a procedure to provide temporary ventilation or opening of doors to provide alternate cooling is potentially cost-beneficial. Table 9-1 has been revised to show this as a potentially cost beneficial SAMA item. The revised table is included as Attachment 1 to this enclosure.

Information on other HVAC to other areas is included in the response to RAI 6.v.

CALLAWAY PLANT UNIT 1
LICENSE RENEWAL APPLICATION

REQUEST FOR ADDITIONAL INFORMATION

Revised SAMA Tables

Table 5-1. List of SAMA Candidates.

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
1	Provide additional DC battery capacity.	Extended DC power availability during an SBO station blackout (SBO).	AC/DC	1
2	Replace lead-acid batteries with fuel cells.	Extended DC power availability during an SBO.	AC/DC	1
3	Add additional battery charger or portable, diesel-driven battery charger to existing DC system.	Improved availability of DC power system.	AC/DC	1
4	Improve DC bus load shedding.	Extended DC power availability during an SBO.	AC/DC	1
5	Provide DC bus cross-ties.	Improved availability of DC power system.	AC/DC	1
6	Provide additional DC power to the 120/240V vital AC system.	Increased availability of the 120 V vital AC bus.	AC/DC	1
7	Add an automatic feature to transfer the 120V vital AC bus from normal to standby power.	Increased availability of the 120 V vital AC bus.	AC/DC	1
8	Increase training on response to loss of two 120V AC buses which causes inadvertent actuation signals.	Improved chances of successful response to loss of two 120V AC buses.	AC/DC	1
9	Provide an additional diesel generator.	Increased availability of on-site emergency AC power.	AC/DC	1
10	Revise procedure to allow bypass of diesel generator trips.	Extended diesel generator operation.	AC/DC	1
11	Improve 4.16-kV bus cross-tie ability.	Increased availability of on-site AC power.	AC/DC	1
12	Create AC power cross-tie capability with other unit (multi-unit site)	Increased availability of on-site AC power.	AC/DC	1
13	Install an additional, buried off-site power source.	Reduced probability of loss of off-site power.	AC/DC	1
14	Install a gas turbine generator.	Increased availability of on-site AC power.	AC/DC	1
15	Install tornado protection on gas turbine generator.	Increased availability of on-site AC power.	AC/DC	1
16	Improve uninterruptible power supplies.	Increased availability of power supplies supporting front-line equipment.	AC/DC	1
17	Create a cross-tie for diesel fuel oil (multi-unit site).	Increased diesel generator availability.	AC/DC	1
18	Develop procedures for replenishing diesel fuel oil.	Increased diesel generator availability.	AC/DC	1
19	Use fire water system as a backup source for diesel cooling.	Increased diesel generator availability.	AC/DC	1
20	Add a new backup source of diesel cooling.	Increased diesel generator availability.	AC/DC	1
21	Develop procedures to repair or replace failed 4 KV breakers.	Increased probability of recovery from failure of breakers that transfer 4.16 kV non-emergency buses from unit station service transformers.	AC/DC	1
22	In training, emphasize steps in recovery of off-site power after an SBO.	Reduced human error probability during off-site power recovery.	AC/DC	1
23	Develop a severe weather conditions procedure.	Improved off-site power recovery following external weather-related events.	AC/DC	1
24	Bury off-site power lines.	Improved off-site power reliability during severe weather.	AC/DC	1
25	Install an independent active or passive high pressure injection system.	Improved prevention of core melt sequences.	Core Cooling	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
26	Provide an additional high pressure injection pump with independent diesel.	Reduced frequency of core melt from small LOCA and SBO sequences.	Core Cooling	1
27	Revise procedure to allow operators to inhibit automatic vessel depressurization in non-ATWS scenarios.	Extended HPCI and RCIC operation.	Core Cooling	1
28	Add a diverse low pressure injection system.	Improved injection capability.	Core Cooling	1
29	Provide capability for alternate injection via diesel-driven fire pump.	Improved injection capability.	Core Cooling	1
30	Improve ECCS suction strainers.	Enhanced reliability of ECCS suction.	Core Cooling	1
31	Add the ability to manually align emergency core cooling system recirculation.	Enhanced reliability of ECCS suction.	Core Cooling	1
32	Add the ability to automatically align emergency core cooling system to recirculation mode upon refueling water storage tank depletion.	Enhanced reliability of ECCS suction.	Core Cooling	1
33	Provide hardware and procedure to refill the reactor water storage tank once it reaches a specified low level.	Extended reactor water storage tank capacity in the event of a steam generator tube rupture (or other LOCAs challenging RWST capacity).	Core Cooling	1
34	Provide an in-containment reactor water storage tank.	Continuous source of water to the safety injection pumps during a LOCA event, since water released from a breach of the primary system collects in the in-containment reactor water storage tank, and thereby eliminates the need to realign the safety injection pumps for long-term post-LOCA recirculation.	Core Cooling	1
35	Throttle low pressure injection pumps earlier in medium or large-break LOCAs to maintain reactor water storage tank inventory.	Extended reactor water storage tank capacity.	Core Cooling	1
36	Emphasize timely recirculation alignment in operator training.	Reduced human error probability associated with recirculation failure.	Core Cooling	1
37	Upgrade the chemical and volume control system to mitigate small LOCAs.	For a plant like the Westinghouse AP600, where the chemical and volume control system cannot mitigate a small LOCA, an upgrade would decrease the frequency of core damage.	Core Cooling	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
38	Change the in-containment reactor water storage tank suction from four check valves to two check and two air-operated valves.	Reduced common mode failure of injection paths.	Core Cooling	1
39	Replace two of the four electric safety injection pumps with diesel-powered pumps.	Reduced common cause failure of the safety injection system. This SAMA was originally intended for the Westinghouse-CE System 80+, which has four trains of safety injection. However, the intent of this SAMA is to provide diversity within the high- and l	Core Cooling	1
40	Provide capability for remote, manual operation of secondary side pilot-operated relief valves in a station blackout.	Improved chance of successful operation during station blackout events in which high area temperatures may be encountered (no ventilation to main steam areas).	Core Cooling	1
41	Create a reactor coolant depressurization system.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	Core Cooling	1
42	Make procedure changes for reactor coolant system depressurization.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	Core Cooling	1
43	Add redundant DC control power for SW pumps.	Increased availability of SW.	Cooling Water	1
44	Replace ECCS pump motors with air-cooled motors.	Elimination of ECCS dependency on component cooling system.	Cooling Water	1
45	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	Reduced frequency of loss of component cooling water and service water.	Cooling Water	1
46	Add a service water pump.	Increased availability of cooling water.	Cooling Water	1
47	Enhance the screen wash system.	Reduced potential for loss of SW due to clogging of screens.	Cooling Water	1
48	Cap downstream piping of normally closed component cooling water drain and vent valves.	Reduced frequency of loss of component cooling water initiating events, some of which can be attributed to catastrophic failure of one of the many single isolation valves.	Cooling Water	1
49	Enhance loss of component cooling water (or loss of service water) procedures to facilitate stopping the reactor coolant pumps.	Reduced potential for reactor coolant pump seal damage due to pump bearing failure.	Cooling Water	1
50	Enhance loss of component cooling water procedure to underscore the desirability of cooling down the reactor coolant system prior to seal LOCA.	Reduced probability of reactor coolant pump seal failure.	Cooling Water	1
51	Additional training on loss of component cooling water.	Improved success of operator actions after a loss of component cooling water.	Cooling Water	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
52	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.	Reduced effect of loss of component cooling water by providing a means to maintain the charging pump seal injection following a loss of normal cooling water.	Cooling Water	1
53	On loss of essential raw cooling water, proceduralize shedding component cooling water loads to extend the component cooling water heat-up time.	Increased time before loss of component cooling water (and reactor coolant pump seal failure) during loss of essential raw cooling water sequences.	Cooling Water	1
54	Increase charging pump lube oil capacity.	Increased time before charging pump failure due to lube oil overheating in loss of cooling water sequences.	Cooling Water	1
55	Install an independent reactor coolant pump seal injection system, with dedicated diesel.	Reduced frequency of core damage from loss of component cooling water, service water, or station blackout.	Cooling Water	1
56	Install an independent reactor coolant pump seal injection system, without dedicated diesel.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout.	Cooling Water	1
57	Use existing hydro test pump for reactor coolant pump seal injection.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout, unless an alternate power source is used.	Cooling Water	1
58	Install improved reactor coolant pump seals.	Reduced likelihood of reactor coolant pump seal LOCA.	Cooling Water	1
59	Install an additional component cooling water pump.	Reduced likelihood of loss of component cooling water leading to a reactor coolant pump seal LOCA.	Cooling Water	1
60	Prevent makeup pump flow diversion through the relief valves.	Reduced frequency of loss of reactor coolant pump seal cooling if spurious high pressure injection relief valve opening creates a flow diversion large enough to prevent reactor coolant pump seal injection.	Cooling Water	1
61	Change procedures to isolate reactor coolant pump seal return flow on loss of component cooling water, and provide (or enhance) guidance on loss of injection during seal LOCA.	Reduced frequency of core damage due to loss of seal cooling.	Cooling Water	1
62	Implement procedures to stagger high pressure safety injection pump use after a loss of service water.	Extended high pressure injection prior to overheating following a loss of service water.	Cooling Water	1
63	Use fire prevention system pumps as a backup seal injection and high pressure makeup source.	Reduced frequency of reactor coolant pump seal LOCA.	Cooling Water	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie.	Improved ability to cool residual heat removal heat exchangers.	Cooling Water	1
65	Install a digital feed water upgrade.	Reduced chance of loss of main feed water following a plant trip.	Feedwater/ Condensate	1
66	Create ability for emergency connection of existing or new water sources to feedwater and condensate systems.	Increased availability of feedwater.	Feedwater/Condensate	1
67	Install an independent diesel for the condensate storage tank makeup pumps.	Extended inventory in CST during an SBO.	Feedwater/Condensate	1
68	Add a motor-driven feedwater pump.	Increased availability of feedwater.	Feedwater/Condensate	1
69	Install manual isolation valves around auxiliary feedwater turbine-driven steam admission valves.	Reduced dual turbine-driven pump maintenance unavailability.	Feedwater/Condensate	1
70	Install accumulators for turbine-driven auxiliary feedwater pump flow control valves.	Eliminates the need for local manual action to align nitrogen bottles for control air following a loss of off-site power.	Feedwater/Condensate	1
71	Install a new condensate storage tank (auxiliary feedwater storage tank).	Increased availability of the auxiliary feedwater system.	Feedwater/Condensate	1
72	Modify the turbine-driven auxiliary feedwater pump to be self-cooled.	Improved success probability during a station blackout.	Feedwater/Condensate	1
73	Proceduralize local manual operation of auxiliary feedwater system when control power is lost.	Extended auxiliary feedwater availability during a station blackout. Also provides a success path should auxiliary feedwater control power be lost in non-station blackout sequences.	Feedwater/Condensate	1
74	Provide hookup for portable generators to power the turbine-driven auxiliary feedwater pump after station batteries are depleted.	Extended auxiliary feedwater availability.	Feedwater/Condensate	1
75	Use fire water system as a backup for steam generator inventory.	Increased availability of steam generator water supply.	Feedwater/Condensate	1
76	Change failure position of condenser makeup valve if the condenser makeup valve fails open on loss of air or power.	Allows greater inventory for the auxiliary feedwater pumps by preventing condensate storage tank flow diversion to the condenser.	Feedwater/Condensate	1
77	Provide a passive, secondary-side heat-rejection loop consisting of a condenser and heat sink.	Reduced potential for core damage due to loss-of-feedwater events.	Feedwater/Condensate	1
78	Modify the startup feedwater pump so that it can be used as a backup to the emergency feedwater system, including during a station blackout scenario.	Increased reliability of decay heat removal.	Feedwater/Condensate	1
79	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.	Increased probability of successful feed and bleed.	Feedwater/Condensate	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
80	Provide a redundant train or means of ventilation.	Increased availability of components dependent on room cooling.	HVAC	1
81	Add a diesel building high temperature alarm or redundant louver and thermostat.	Improved diagnosis of a loss of diesel building HVAC.	HVAC	1
82	Stage backup fans in switchgear rooms.	Increased availability of ventilation in the event of a loss of switchgear ventilation.	HVAC	1
83	Add a switchgear room high temperature alarm.	Improved diagnosis of a loss of switchgear HVAC.	HVAC	1
84	Create ability to switch emergency feedwater room fan power supply to station batteries in a station blackout.	Continued fan operation in a station blackout.	HVAC	1
85	Provide cross-unit connection of uninterruptible compressed air supply.	Increased ability to vent containment using the hardened vent.	IA/Nitrogen	1
86	Modify procedure to provide ability to align diesel power to more air compressors.	Increased availability of instrument air after a LOOP.	IA/Nitrogen	1
87	Replace service and instrument air compressors with more reliable compressors which have self-contained air cooling by shaft driven fans.	Elimination of instrument air system dependence on service water cooling.	IA/Nitrogen	1
88	Install nitrogen bottles as backup gas supply for safety relief valves.	Extended SRV operation time.	IA/Nitrogen	1
89	Improve SRV and MSIV pneumatic components.	Improved availability of SRVs and MSIVs.	IA/Nitrogen	1
90	Create a reactor cavity flooding system.	Enhanced debris cool ability, reduced core concrete interaction, and increased fission product scrubbing.	Containment Phenomena	1
91	Install a passive containment spray system.	Improved containment spray capability.	Containment Phenomena	1
92	Use the fire water system as a backup source for the containment spray system.	Improved containment spray capability.	Containment Phenomena	1
93	Install an unfiltered, hardened containment vent.	Increased decay heat removal capability for non-ATWS events, without scrubbing released fission products.	Containment Phenomena	1
94	Install a filtered containment vent to remove decay heat. Option 1: Gravel Bed Filter; Option 2: Multiple Venturi Scrubber	Increased decay heat removal capability for non-ATWS events, with scrubbing of released fission products.	Containment Phenomena	1
95	Enhance fire protection system and standby gas treatment system hardware and procedures.	Improved fission product scrubbing in severe accidents.	Containment Phenomena	1
96	Provide post-accident containment inerting capability.	Reduced likelihood of hydrogen and carbon monoxide gas combustion.	Containment Phenomena	1
97	Create a large concrete crucible with heat removal potential to contain molten core debris.	Increased cooling and containment of molten core debris. Molten core debris escaping from the vessel is contained within the crucible and a water cooling mechanism cools the molten core in the crucible, preventing melt-through of the base mat.	Containment Phenomena	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
98	Create a core melt source reduction system.	Increased cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal from the vitrified compound would be facilitated, and concrete attack would not occur.	Containment Phenomena	1
99	Strengthen primary/secondary containment (e.g., add ribbing to containment shell).	Reduced probability of containment over-pressurization.	Containment Phenomena	1
100	Increase depth of the concrete base mat or use an alternate concrete material to ensure melt-through does not occur.	Reduced probability of base mat melt-through.	Containment Phenomena	1
101	Provide a reactor vessel exterior cooling system.	Increased potential to cool a molten core before it causes vessel failure, by submerging the lower head in water.	Containment Phenomena	1
102	Construct a building to be connected to primary/secondary containment and maintained at a vacuum.	Reduced probability of containment over-pressurization.	Containment Phenomena	1
103	Institute simulator training for severe accident scenarios.	Improved arrest of core melt progress and prevention of containment failure.	Containment Phenomena	1
104	Improve leak detection procedures.	Increased piping surveillance to identify leaks prior to complete failure. Improved leak detection would reduce LOCA frequency.	Containment Phenomena	1
105	Delay containment spray actuation after a large LOCA.	Extended reactor water storage tank availability.	Containment Phenomena	1
106	Install automatic containment spray pump header throttle valves.	Extended time over which water remains in the reactor water storage tank, when full containment spray flow is not needed.	Containment Phenomena	1
107	Install a redundant containment spray system.	Increased containment heat removal ability.	Containment Phenomena	1
108	Install an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel.	Reduced hydrogen detonation potential.	Containment Phenomena	1
109	Install a passive hydrogen control system.	Reduced hydrogen detonation potential.	Containment Phenomena	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
110	Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure.	Reduced probability of containment failure.	Containment Phenomena	1
111	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.	Reduced ISLOCA frequency.	Containment Bypass	1
112	Add redundant and diverse limit switches to each containment isolation valve.	Reduced frequency of containment isolation failure and ISLOCAs.	Containment Bypass	1
113	Increase leak testing of valves in ISLOCA paths.	Reduced ISLOCA frequency.	Containment Bypass	1
114	Install self-actuating containment isolation valves.	Reduced frequency of isolation failure.	Containment Bypass	1
115	Locate residual heat removal (RHR) inside containment	Reduced frequency of ISLOCA outside containment.	Containment Bypass	1
116	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Scrubbed ISLOCA releases.	Containment Bypass	1
117	Revise EOPs to improve ISLOCA identification.	Increased likelihood that LOCAs outside containment are identified as such. A plant had a scenario in which an RHR ISLOCA could direct initial leakage back to the pressurizer relief tank, giving indication that the LOCA was inside containment.	Containment Bypass	1
118	Improve operator training on ISLOCA coping.	Decreased ISLOCA consequences.	Containment Bypass	1
119	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage.	Reduced frequency of steam generator tube ruptures.	Containment Bypass	1
120	Replace steam generators with a new design.	Reduced frequency of steam generator tube ruptures.	Containment Bypass	1
121	Increase the pressure capacity of the secondary side so that a steam generator tube rupture would not cause the relief valves to lift.	Eliminates release pathway to the environment following a steam generator tube rupture.	Containment Bypass	1
122	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture	Enhanced depressurization capabilities during steam generator tube rupture.	Containment Bypass	1
123	Proceduralize use of pressurizer vent valves during steam generator tube rupture sequences.	Backup method to using pressurizer sprays to reduce primary system pressure following a steam generator tube rupture.	Containment Bypass	1
124	Provide improved instrumentation to detect steam generator tube ruptures, such as Nitrogen-16 monitors).	Improved mitigation of steam generator tube ruptures.	Containment Bypass	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
125	Route the discharge from the main steam safety valves through a structure where a water spray would condense the steam and remove most of the fission products.	Reduced consequences of a steam generator tube rupture.	Containment Bypass	1
126	Install a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources	Reduced consequences of a steam generator tube rupture.	Containment Bypass	1
127	Revise emergency operating procedures to direct isolation of a faulted steam generator.	Reduced consequences of a steam generator tube rupture.	Containment Bypass	1
128	Direct steam generator flooding after a steam generator tube rupture, prior to core damage.	Improved scrubbing of steam generator tube rupture releases.	Containment Bypass	1
129	Vent main steam safety valves in containment.	Reduced consequences of a steam generator tube rupture.	Containment Bypass	1
130	Add an independent boron injection system.	Improved availability of boron injection during ATWS.	ATWS	1
131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Improved equipment availability after an ATWS.	ATWS	1
132	Provide an additional control system for rod insertion (e.g., AMSAC).	Improved redundancy and reduced ATWS frequency.	ATWS	1
133	Install an ATWS sized filtered containment vent to remove decay heat.	Increased ability to remove reactor heat from ATWS events.	ATWS	1
134	Revise procedure to bypass MSIV isolation in turbine trip ATWS scenarios.	Affords operators more time to perform actions. Discharge of a substantial fraction of steam to the main condenser (i.e., as opposed to into the primary containment) affords the operator more time to perform actions (e.g., SLC injection, lower water level, depressurize RPV) than if the main condenser was unavailable, resulting in lower human error probabilities.	ATWS	1
135	Revise procedure to allow override of low pressure core injection during an ATWS event.	Allows immediate control of low pressure core injection. On failure of high pressure core injection and condensate, some plants direct reactor depressurization followed by five minutes of automatic low pressure core injection.	ATWS	1
136	Install motor generator set trip breakers in control room.	Reduced frequency of core damage due to an ATWS.	ATWS	1
137	Provide capability to remove power from the bus powering the control rods.	Decreased time required to insert control rods if the reactor trip breakers fail (during a loss of feedwater ATWS which has rapid pressure excursion).	ATWS	1
138	Improve inspection of rubber expansion joints on main condenser.	Reduced frequency of internal flooding due to failure of circulating water system expansion joints.	Internal Flooding	1
139	Modify swing direction of doors separating turbine building basement from areas containing safeguards equipment.	Prevents flood propagation.	Internal Flooding	1

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
140	Increase seismic ruggedness of plant components.	Increased availability of necessary plant equipment during and after seismic events.	Seismic Risk	1
141	Provide additional restraints for CO2 tanks.	Increased availability of fire protection given a seismic event.	Seismic Risk	1
142	Replace mercury switches in fire protection system.	Decreased probability of spurious fire suppression system actuation.	Fire Risk	1
143	Upgrade fire compartment barriers.	Decreased consequences of a fire.	Fire Risk	1
144	Install additional transfer and isolation switches.	Reduced number of spurious actuations during a fire.	Fire Risk	1
145	Enhance fire brigade awareness.	Decreased consequences of a fire.	Fire Risk	1
146	Enhance control of combustibles and ignition sources.	Decreased fire frequency and consequences.	Fire Risk	1
147	Install digital large break LOCA protection system.	Reduced probability of a large break LOCA (a leak before break).	Other	1
148	Enhance procedures to mitigate large break LOCA.	Reduced consequences of a large break LOCA.	Other	1
149	Install computer aided instrumentation system to assist the operator in assessing post-accident plant status.	Improved prevention of core melt sequences by making operator actions more reliable.	Other	1
150	Improve maintenance procedures.	Improved prevention of core melt sequences by increasing reliability of important equipment.	Other	1
151	Increase training and operating experience feedback to improve operator response.	Improved likelihood of success of operator actions taken in response to abnormal conditions.	Other	1
152	Develop procedures for transportation and nearby facility accidents.	Reduced consequences of transportation and nearby facility accidents.	Other	1
153	Install secondary side guard pipes up to the main steam isolation valves.	Prevents secondary side depressurization should a steam line break occur upstream of the main steam isolation valves. Also guards against or prevents consequential multiple steam generator tube ruptures following a main steam line break event.	Other	1
154	Mount or anchor the MCCs to the respective building walls.	Reduces failure probability of MCCs during an earthquake	IPEEE - Seismic	B
155	Install shear pins (or strength bolts) in the AFW pumps.	Takes up the shear load on the pump and/or driver during an earthquake.	IPEEE - Seismic	B
156	Mount all fire extinguishers within their UL Standard required drop height and remove hand-held fire extinguishers from Containment during normal operation.	Reduces the potential for the fire extinguishers to fall during an earthquake and potentially fracturing upon impact with the floor or another object.	IPEEE - Seismic	B

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
157	Identify and remove unsecured equipment near areas that contain relays that actuate, so area is kept clear.	Ensures direct access to areas such as Load Shedding and Emergency Load Sequencing (LSELS) and Engineered Safety Feature Actuation System (ESFAS) cabinets. Unsecured equipment (e.g., carts, filing cabinets, and test equipment) in these areas could result	IPEEE – Seismic	B
158	Properly position chain hoists that facilitate maintenance on pumps within pump rooms and institute a training program to ensure that the hoists are properly positioned when not in use.	Improper positioning of hoists reduces the availability due to moving during an earthquake and having chainfalls impacting pump oil bubblers or other soft targets resulting in failure of the pumps.	IPEEE – Seismic	B
159	Secure floor grating to prevent damage to sensing lines due to differential building motion.	Prevent sensing lines that pass through the grating from being damaged.	IPEEE – Seismic	B
160	Modifications to lessen impact of internal flooding path through Control Building dumbwaiter.	Lower impact of flood that propagates through the dumbwaiter	Internal Flooding	D
161	Improvements to PORV performance that will lower the probability of failure to open.	Decrease in risk due to PORV failing to open.	Core Cooling	E
162	Install a large volume EDG fuel oil tank at an elevation greater than the EDG fuel oil day tanks.	Allows transfer of EDF fuel oil to the EDG day tanks on failure of the fuel oil transfer pumps.	AC/DC	C
163	Improve feedwater check valve reliability to reduce probability of failure to open.	Lower risk due to failures in which feedwater check valves fail to open and allow feeding of the steam generators.	Cooling Water	E
164	Provide the capability to power the normal service water pumps from AEPS.	Provide backup to ESW in conditions with power only available from AEPS.	Cooling Water	D
165	Purchase or manufacture a "gagging device" that could be used to close a stuck open steam generator relief valve for a SGTR event prior to core damage.	Reduce the amount of radioactive material release to the atmosphere in a SGTR event with core damage.	SGTR	C
166	Installation of high temperature qualified RCP seal O-rings.	Lower potential for RCP seal leakage.	RCP Seal LOCA	A
167	Addition of procedural guidance to re-establish normal service water should essential service water fail.	Provide back-up pumps for UHS cooling.	Cooling Water	A
168	Addition of procedural guidance for running charging and safety injection pumps without component cooling water	Allow use of pumps following loss of component cooling water.	Cooling Water	A
169	Addition of procedural guidance to verify RHR pump room cooling at switchover to ECCS recirculation phase.	Verifying that support system for RHR pumps is in service to allow continued operation of RHR pumps.	HVAC	A
170	Modifications to add controls in the main control room to allow remote operation of nearby diesel generator farm and alignment/connection to the plant vital electrical busses.	Faster ability to provide power to the plant electrical busses from the offsite diesel generator farm.	AC Power	C

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
171	Increase the size of the RWST or otherwise improve the availability of the RWST	Ensure a supply of makeup water is available from the RWST.	Core Cooling	E
172	Addition of procedural guidance and the required hardware to enable the operators to feed one or more steam generators with a diesel driven firewater pump.	Provide a backup to turbine driven auxiliary feedwater.	Feedwater	A
173	Addition of a black start combustion turbine generator.	A redundant source of AC Power that could be used in station blackout events.	AC Power	A
174	Addition of a black-start engine-generator to provide AC Power during a station blackout	Ability to power a 125VDC battery charger and a charging pump. Powering the battery charger would permit operation of the TDAFP without recovering AC power. Powering a charging pump could provide RCP seal injection and preclude a RCP seal LOCA during a station blackout.	AC Power	A
175	Replacement of the positive displacement charging pump with a third centrifugal charging pump.	Provide another source for RCP seal cooling, RCS makeup, and pumped flow for feed and bleed.	Cooling Water	A
176	Provide control modifications to bypass feedwater isolation in order to restore main feedwater.	Allow faster and more reliable bypass of the main feedwater isolation signal in order to restore main feedwater to the steam generators should auxiliary feedwater fail.	Feedwater	A
177	Procedural and hardware modifications to reduce core damage risk due to internal flooding.	The IPE identified a need to form a task force to identify and evaluate potential procedural and hardware modifications aimed at reducing the risk due to internal flooding.	Flooding	A
178	Improvements to UHS cooling tower electrical room HVAC.	Improve availability or mitigate loss of HVAC.	HVAC	E
179	Modify procedures such that the water loop seals in the RCS cold legs are not cleared following core damage.	Prevents possible thermally induced steam generator tube rupture following core damage.	Containment Bypass	C
180	Install lower amperage fuses for various 14 AWG control circuits in the MCR. The majority of the modification centers around the trip circuit fuses on NB, NG, PA, PB, and PG system breakers.	Reduced fire risk.	Fire Risk	F
181	Install redundant fuses and isolation switches for MCR evacuation procedure OTO-ZZ-00001.	Reduced fire risk.	Fire Risk	F
182	To protect against multiple spurious operation scenarios, cable runs will be changed to run a single wire in a protected metal jacket such that spurious valve opening due to a hot short affecting the valve control circuit is eliminated for the fire area. This modification will be implemented in multiple fire areas.	Reduced fire risk.	Fire Risk	F

Table 5-1. List of SAMA Candidates (Continued).

Callaway SAMA Number	Potential Improvement	Discussion	Focus of SAMA	Source
183	Quick response sprinkler heads in cable chases A-11, C-30, and C-31 will be modified to be in accordance with the applicable requirements of NFPA 13-1976 edition.	Reduced fire risk.	Fire Risk	F
184	Improvements in the reliability of the Steam Line Isolation automatic signal.	More reliable main steam line isolation.	Containment Isolation	E
185	Automate initiation of CCW flow to the RHR heat exchangers.	More reliable than manual initiation of flow to RRHR HX.	Cooling Water	E
186	Develop a procedure and obtain equipment to provide a temporary hookup of fire water as a replacement for ESW	Backup cooling water if ESW/SW is lost	Cooling Water	D

Note 1: The source references are:
- 1 NEI 05-01 (Reference 19)
- A IPE (Reference 28)
- B IPEEE (Reference 29)
- C Recent industry SAMA submittals (Wolf Creek, South Texas, Diablo Canyon, Seabrook)
- D Expert panel convened to review SAMA analysis or other plant personnel
- E PRA importance list review
- F Callaway NFPA 805 License Amendment Request

6.0 PHASE I ANALYSIS

A preliminary screening of the complete list of SAMA candidates was performed to limit the number of SAMAs for which detailed analysis in Phase II was necessary. The screening criteria used in the Phase I analysis are described below.

- Screening Criterion A - Not Applicable: If a SAMA candidate did not apply to the Callaway Unit 1 plant design, it was not retained.
- Screening Criterion B - Already Implemented or Intent Met: If a SAMA candidate had already been implemented at the Callaway Plant or its intended benefit already achieved by other means, it was not retained.
- Screening Criterion C - Combined: If a SAMA candidate was similar in nature and could be combined with another SAMA candidate to develop a more comprehensive or plant-specific SAMA candidate, only the combined SAMA candidate was retained.
- Screening Criterion D - Excessive Implementation Cost: If a SAMA required extensive changes that will obviously exceed the maximum benefit (Section 4.5), even without an implementation cost estimate, it was not retained.
- Screening Criterion E - Very Low Benefit: If a SAMA from an industry document was related to a non-risk significant system for which change in reliability is known to have negligible impact on the risk profile, it was not retained. (No SAMAs were screened using this criterion.)

Table 6-1 presents the list of Phase I SAMA candidates and provides the disposition of each candidate along with the applicable screening criterion associated with each candidate. Those candidates that have not been screened by application of these criteria are evaluated further in the Phase II analysis (Section 7). It can be seen from this table that 107 SAMAs were screened from the analysis during Phase 1 and that 64 SAMAs passed into the next phase of the analysis.

Table 6-1. Callaway Plant Phase I SAMA Analysis

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
12	Create AC power cross-tie capability with other unit (multi-unit site)	Increased availability of on-site AC power.	Yes	A - Not Applicable	Callaway is a single unit site.
17	Create a cross-tie for diesel fuel oil (multi-unit site).	Increased diesel generator availability.	Yes	A - Not Applicable	Callaway is a single unit site.
27	Revise procedure to allow operators to inhibit automatic vessel depressurization in non-ATWS scenarios.	Extended HPCI and RCIC operation.	Yes	A - Not Applicable	BWR item.
34	Provide an in-containment reactor water storage tank.	Continuous source of water to the safety injection pumps during a LOCA event, since water released from a breach of the primary system collects in the in-containment reactor water storage tank, and thereby eliminates the need to realign the safety injection pumps for long-term post-LOCA recirculation.	Yes	A - Not Applicable	Not applicable for existing designs. Insufficient room inside primary containment.
35	Throttle low pressure injection pumps earlier in medium or large-break LOCAs to maintain reactor water storage tank inventory.	Extended reactor water storage tank capacity.	Yes	A - Not Applicable	Per the Callaway safety analysis, this is an undesirable action. The Callaway safety analysis and design calls for injection of the RWST to inside the containment as soon as possible.
38	Change the in-containment reactor water storage tank suction from four check valves to two check and two air-operated valves.	Reduced common mode failure of injection paths.	Yes	A - Not Applicable	Callaway does not have an in-containment RWST with this valve arrangement.
47	Enhance the screen wash system.	Reduced potential for loss of SW due to clogging of screens.	Yes	A - Not Applicable	Plant uses Ultimate Heat Sink pond for cooling. UHS sized for 30 days without make-up. River intake is only used for make-up to the UHS.
52	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.	Reduced effect of loss of component cooling water by providing a means to maintain the charging pump seal injection following a loss of normal cooling water.	Yes	A - Not Applicable	Charging pump seals do not require external cooling, they are cooled by the process fluid.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
57	Use existing hydro test pump for reactor coolant pump seal injection.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout, unless an alternate power source is used.	Yes	A - Not Applicable	Callaway does not have a permanently installed hydro test pump. Timing considerations prevent credit for hookup of temporary pump.
63	Use fire prevention system pumps as a backup seal injection and high pressure makeup source.	Reduced frequency of reactor coolant pump seal LOCA.	Yes	A - Not Applicable	Existing fire protection system pumps do not have sufficient discharge head to use as high pressure makeup source.
69	Install manual isolation valves around auxiliary feedwater turbine-driven steam admission valves.	Reduced dual turbine-driven pump maintenance unavailability.	Yes	A - Not Applicable	Callaway does not have dual turbine AFW pump.
85	Provide cross-unit connection of uninterruptible compressed air supply.	Increased ability to vent containment using the hardened vent.	Yes	A - Not Applicable	N/A, single unit.
95	Enhance fire protection system and standby gas treatment system hardware and procedures.	Improved fission product scrubbing in severe accidents.	Yes	A - Not Applicable	Standby gas treatment system is BWR item.
105	Delay containment spray actuation after a large LOCA.	Extended reactor water storage tank availability.	Yes	A - Not Applicable	Per the Callaway safety analysis, this is an undesirable action. The Callaway safety analysis and design calls for injection of the RWST to inside the containment as soon as possible.
106	Install automatic containment spray pump header throttle valves.	Extended time over which water remains in the reactor water storage tank, when full containment spray flow is not needed.	Yes	A - Not Applicable	Per the Callaway safety analysis, this is an undesirable action. The Callaway safety analysis and design calls for injection of the RWST to inside the containment as soon as possible.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
134	Revise procedure to bypass MSIV isolation in turbine trip ATWS scenarios.	Affords operators more time to perform actions. Discharge of a substantial fraction of steam to the main condenser (i.e., as opposed to into the primary containment) affords the operator more time to perform actions (e.g., SLC injection, lower water level, depressurize RPV) than if the main condenser was unavailable, resulting in lower human error probabilities.	Yes	A - Not Applicable	Specific to BWRs.
135	Revise procedure to allow override of low pressure core injection during an ATWS event.	Allows immediate control of low pressure core injection. On failure of high pressure core injection and condensate, some plants direct reactor depressurization followed by five minutes of automatic low pressure core injection.	Yes	A - Not Applicable	Based on description, this is a BWR item.
138	Improve inspection of rubber expansion joints on main condenser.	Reduced frequency of internal flooding due to failure of circulating water system expansion joints.	Yes	A - Not Applicable	No risk significant flooding sources identified in the turbine building.
139	Modify swing direction of doors separating turbine building basement from areas containing safeguards equipment.	Prevents flood propagation.	Yes	A - Not Applicable	Flooding analysis did not indicate any flooding issues related to the direction of door swing.
142	Replace mercury switches in fire protection system.	Decreased probability of spurious fire suppression system actuation.	Yes	A - Not Applicable	No mercury switches in the fire protection system.
143	Upgrade fire compartment barriers.	Decreased consequences of a fire.	Yes	A - Not Applicable	Fire analysis did not identify any issues related to fire barriers. NFPA 805 Fire Protection Program is in progress, any issues identified by that project will be handled by the NFPA 805 program.
152	Develop procedures for transportation and nearby facility accidents.	Reduced consequences of transportation and nearby facility accidents.	Yes	A - Not Applicable	IPEEE determined that there are no transportation routes or nearby facilities that could cause concern.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
165	Purchase or manufacture a "gagging device" that could be used to close a stuck open steam generator relief valve for a SGTR event prior to core damage.	Reduce the amount of radioactive material release to the atmosphere in a SGTR event with core damage.	Yes	A - Not Applicable	Callaway does not have the ability to isolate the steam generator from the RCS loop. The amount of force required to close a stuck open atmospheric steam dump valve would likely not be successful and would result in further damage to the valve.
3	Add additional battery charger or portable, diesel-driven battery charger to existing DC system.	Improved availability of DC power system.	Yes	B - Intent Met	Current configuration is two spare battery chargers for the instrument buses. The spare can carry one bus. One feeds A/B, the other feeds C/D trains. Also Emergency Coordinator Supplemental Guidelines, Attachment N, "Temporary Power to NK Swing Charger
4	Improve DC bus load shedding.	Extended DC power availability during an SBO.	Yes	B - Intent Met	DC load shedding is conducted.
6	Provide additional DC power to the 120/240V vital AC system.	Increased availability of the 120 V vital AC bus.	Yes	B - Intent Met	Procedures in place to provide temporary power to DC Chargers which can power vital AC system.
7	Add an automatic feature to transfer the 120V vital AC bus from normal to standby power.	Increased availability of the 120 V vital AC bus.	Yes	B - Intent Met	On loss of DC or inverter, the UPS static switch automatically transfers to AC power through a constant voltage transformer. An additional backup AC source is available, but must be closed manually.
8	Increase training on response to loss of two 120V AC buses which causes inadvertent actuation signals.	Improved chances of successful response to loss of two 120V AC buses.	Yes	B - Intent Met	Typical response training in place.
9	Provide an additional diesel generator.	Increased availability of on-site emergency AC power.	Yes	B - Intent Met	Alternate Emergency Power System installed.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
10	Revise procedure to allow bypass of diesel generator trips.	Extended diesel generator operation.	Yes	B - Intent Met	Bypass of non-vital diesel generator trips were in original design for Callaway.
13	Install an additional, buried off-site power source.	Reduced probability of loss of off-site power.	Yes	B - Intent Met	AEPS installed with buried power lines.
14	Install a gas turbine generator.	Increased availability of on-site AC power.	Yes	B - Intent Met	Alternate Emergency Power System installed.
16	Improve uninterruptible power supplies.	Increased availability of power supplies supporting front-line equipment.	Yes	B - Intent Met	Replaced to add static switch and upgrade to newer design.
18	Develop procedures for replenishing diesel fuel oil.	Increased diesel generator availability.	Yes	B - Intent Met	EOP Addenda direct ordering fuel oil.
19	Use fire water system as a backup source for diesel cooling.	Increased diesel generator availability.	Yes	B - Intent Met	Procedures exist for cooling EDG with fire water.
20	Add a new backup source of diesel cooling.	Increased diesel generator availability.	Yes	B - Intent Met	Procedure exists for backup diesel cooling.
21	Develop procedures to repair or replace failed 4 KV breakers.	Increased probability of recovery from failure of breakers that transfer 4.16 kV non-emergency buses from unit station service transformers.	Yes	B - Intent Met	Spares exist and procedures exist.
22	In training, emphasize steps in recovery of off-site power after an SBO.	Reduced human error probability during off-site power recovery.	Yes	B - Intent Met	Recovery stressed in training.
23	Develop a severe weather conditions procedure.	Improved off-site power recovery following external weather-related events.	Yes	B - Intent Met	Severe weather condition procedure in place.
30	Improve ECCS suction strainers.	Enhanced reliability of ECCS suction.	Yes	B - Intent Met	Callaway has implemented a containment sump modification that now uses state-of-the-art strainers to address the industry's concerns on blockage from debris. This modification occurred over two outages in 2007 and 2008.
31	Add the ability to manually align emergency core cooling system recirculation.	Enhanced reliability of ECCS suction.	Yes	B - Intent Met	Current alignment capabilities are half and half (manual/automatic).

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
32	Add the ability to automatically align emergency core cooling system to recirculation mode upon refueling water storage tank depletion.	Enhanced reliability of ECCS suction.	Yes	B - Intent Met	Current alignment capabilities are half and half (manual/automatic).
33	Provide hardware and procedure to refill the reactor water storage tank once it reaches a specified low level.	Extended reactor water storage tank capacity in the event of a steam generator tube rupture (or other LOCAs challenging RWST capacity).	Yes	B - Intent Met	Addressed in SAMGs and the EC Supplemental Guideline.
36	Emphasize timely recirculation alignment in operator training.	Reduced human error probability associated with recirculation failure.	Yes	B - Intent Met	Current alignment capabilities are half and half (manual/automatic). Swap to recirculation is stressed in operator training.
37	Upgrade the chemical and volume control system to mitigate small LOCAs.	For a plant like the Westinghouse AP600, where the chemical and volume control system cannot mitigate a small LOCA, an upgrade would decrease the frequency of core damage.	Yes	B - Intent Met	CVCS system is capable of mitigating small LOCA.
40	Provide capability for remote, manual operation of secondary side pilot-operated relief valves in a station blackout.	Improved chance of successful operation during station blackout events in which high area temperatures may be encountered (no ventilation to main stream areas).	Yes	B - Intent Met	Remote Operation of Atmospheric Steam Dumps (ASDs) is possible. Equipment Operators trained and Operator Aid posted.
42	Make procedure changes for reactor coolant system depressurization.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	Yes	B - Intent Met	Multiple depressurization methods are in place.
44	Replace ECCS pump motors with air-cooled motors.	Elimination of ECCS dependency on component cooling system.	Yes	B - Intent Met	Current ECCS pump motors are air-cooled. Additionally the plant OTN procedures allow for alternate trains to supply cooling.
45	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	Reduced frequency of loss of component cooling water and service water.	Yes	B - Intent Met	Can use service water as backup to ESW.
48	Cap downstream piping of normally closed component cooling water drain and vent valves.	Reduced frequency of loss of component cooling water initiating events, some of which can be attributed to catastrophic failure of one of the many single isolation valves.	Yes	B - Intent Met	Vents & drains capped.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
49	Enhance loss of component cooling water (or loss of service water) procedures to facilitate stopping the reactor coolant pumps.	Reduced potential for reactor coolant pump seal damage due to pump bearing failure.	Yes	B - Intent Met	CCW is cooled by ESW. Currently authorized to run 10 minutes.
50	Enhance loss of component cooling water procedure to underscore the desirability of cooling down the reactor coolant system prior to seal LOCA.	Reduced probability of reactor coolant pump seal failure.	Yes	B - Intent Met	Procedures include direction to cool down to minimize impact of RCP seal LOCA.
51	Additional training on loss of component cooling water.	Improved success of operator actions after a loss of component cooling water.	Yes	B - Intent Met	Training is conducted for Loss of CCW.
53	On loss of essential raw cooling water, proceduralize shedding component cooling water loads to extend the component cooling water heat-up time.	Increased time before loss of component cooling water (and reactor coolant pump seal failure) during loss of essential raw cooling water sequences.	Yes	B - Intent Met	Most non-safety loads have been removed from the system. Non-safety loop is automatically isolated on safety injection signal.
60	Prevent makeup pump flow diversion through the relief valves.	Reduced frequency of loss of reactor coolant pump seal cooling if spurious high pressure injection relief valve opening creates a flow diversion large enough to prevent reactor coolant pump seal injection.	Yes	B - Intent Met	Current configuration does not have a relief valve.
61	Change procedures to isolate reactor coolant pump seal return flow on loss of component cooling water, and provide (or enhance) guidance on loss of injection during seal LOCA.	Reduced frequency of core damage due to loss of seal cooling.	Yes	B - Intent Met	Procedure exist
62	Implement procedures to stagger high pressure safety injection pump use after a loss of service water.	Extended high pressure injection prior to overheating following a loss of service water.	Yes	B - Intent Met	Procedure currently in place to stagger use of HPSI.
66	Create ability for emergency connection of existing or new water sources to feedwater and condensate systems.	Increased availability of feedwater.	Yes	B - Intent Met	Procedures exist.
67	Install an independent diesel for the condensate storage tank makeup pumps.	Extended inventory in CST during an SBO.	Yes	B - Intent Met	Procedures do exist for make-up to CST from fire water and for supplying fire water directly to the TDAFW pump.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
68	Add a motor-driven feedwater pump.	Increased availability of feedwater.	Yes	B - Intent Met	Non-Safety Auxiliary Feedwater Pump installed.
70	Install accumulators for turbine-driven auxiliary feedwater pump flow control valves.	Eliminates the need for local manual action to align nitrogen bottles for control air following a loss of off-site power.	Yes	B - Intent Met	Currently have nitrogen accumulators.
72	Modify the turbine-driven auxiliary feedwater pump to be self-cooled.	Improved success probability during a station blackout.	Yes	B - Intent Met	Turbine-driven auxiliary feedwater pump is self-cooled.
73	Proceduralize local manual operation of auxiliary feedwater system when control power is lost.	Extended auxiliary feedwater availability during a station blackout. Also provides a success path should auxiliary feedwater control power be lost in non-station blackout sequences.	Yes	B - Intent Met	Procedures exist.
74	Provide hookup for portable generators to power the turbine-driven auxiliary feedwater pump after station batteries are depleted.	Extended auxiliary feedwater availability.	Yes	B - Intent Met	Procedures exist, hardware on site.
75	Use fire water system as a backup for steam generator inventory.	Increased availability of steam generator water supply.	Yes	B - Intent Met	Equipment staged at CST for makeup. See operator aids. Procedural guidance exists.
76	Change failure position of condenser makeup valve if the condenser makeup valve fails open on loss of air or power.	Allows greater inventory for the auxiliary feedwater pumps by preventing condensate storage tank flow diversion to the condenser.	Yes	B - Intent Met	Valve currently fails closed.
78	Modify the startup feedwater pump so that it can be used as a backup to the emergency feedwater system, including during a station blackout scenario.	Increased reliability of decay heat removal.	Yes	B - Intent Met	Non-Safety Auxiliary Feedwater Pump gets power from Alternate Emergency Power System.
81	Add a diesel building high temperature alarm or redundant louver and thermostat.	Improved diagnosis of a loss of diesel building HVAC.	Yes	B - Intent Met	Computer points for monitoring diesel room temperatures.
82	Stage backup fans in switchgear rooms.	Increased availability of ventilation in the event of a loss of switchgear ventilation.	Yes	B - Intent Met	Procedures include instructions for opening doors to provide alternate cooling capability.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
83	Add a switchgear room high temperature alarm.	Improved diagnosis of a loss of switchgear HVAC.	Yes	B - Intent Met	Plant Process Computer has alarming computer points for switchgear room temperature.
84	Create ability to switch emergency feedwater room fan power supply to station batteries in a station blackout.	Continued fan operation in a station blackout.	Yes	B - Intent Met	Procedure currently in place to switch fan power supply.
86	Modify procedure to provide ability to align diesel power to more air compressors.	Increased availability of instrument air after a LOOP.	Yes	B - Intent Met	Currently have 3 air compressors (service air). A/B compressors are powered off the emergency buses (cooled from essential service lines). Compressors are initially load shed, but procedure direct operators to override and place compressor in service.
88	Install nitrogen bottles as backup gas supply for safety relief valves.	Extended SRV operation time.	Yes	B - Intent Met	Current configuration includes nitrogen bottles as backup gas supply.
89	Improve SRV and MSIV pneumatic components.	Improved availability of SRVs and MSIVs.	Yes	B - Intent Met	MSIV actuators changed to process fluid actuated. Modification installed to relocate Atmospheric Steam Dump valve controllers.
90	Create a reactor cavity flooding system.	Enhanced debris cool ability, reduced core concrete interaction, and increased fission product scrubbing.	Yes	B - Intent Met	Procedures exist
92	Use the fire water system as a backup source for the containment spray system.	Improved containment spray capability.	Yes	B - Intent Met	Procedures exist
101	Provide a reactor vessel exterior cooling system.	Increased potential to cool a molten core before it causes vessel failure, by submerging the lower head in water.	Yes	B - Intent Met	Procedures exist.
103	Institute simulator training for severe accident scenarios.	Improved arrest of core melt progress and prevention of containment failure.	Yes	B - Intent Met	Operators are trained on the SAMG that the operators must implement.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
117	Revise EOPs to improve ISLOCA identification.	Increased likelihood that LOCAs outside containment are identified as such. A plant had a scenario in which an RHR ISLOCA could direct initial leakage back to the pressurizer relief tank, giving indication that the LOCA was inside containment.	Yes	B - Intent Met	Current EOPs address ISLOCA identification.
118	Improve operator training on ISLOCA coping.	Decreased ISLOCA consequences.	Yes	B - Intent Met	Current procedure training addresses ISLOCA identification.
120	Replace steam generators with a new design.	Reduced frequency of steam generator tube ruptures.	Yes	B - Intent Met	Replaced during the fall of 2005 (newer design) which consist of 72,000 sq. ft. per generator.
123	Proceduralize use of pressurizer vent valves during steam generator tube rupture sequences.	Backup method to using pressurizer sprays to reduce primary system pressure following a steam generator tube rupture.	Yes	B - Intent Met	Procedure currently in place.
124	Provide improved instrumentation to detect steam generator tube ruptures, such as Nitrogen-16 monitors).	Improved mitigation of steam generator tube ruptures.	Yes	B - Intent Met	Modification installed to improve operation of N16 detectors.
127	Revise emergency operating procedures to direct isolation of a faulted steam generator.	Reduced consequences of a steam generator tube rupture.	Yes	B - Intent Met	EOP currently in place.
128	Direct steam generator flooding after a steam generator tube rupture, prior to core damage.	Improved scrubbing of steam generator tube rupture releases.	Yes	B - Intent Met	Procedures direct that steam generator level be maintained above the tubes.
132	Provide an additional control system for rod insertion (e.g., AMSAC).	Improved redundancy and reduced ATWS frequency.	Yes	B - Intent Met	Currently have AMSAC.
137	Provide capability to remove power from the bus powering the control rods.	Decreased time required to insert control rods if the reactor trip breakers fail (during a loss of feedwater ATWS which has rapid pressure excursion).	Yes	B - Intent Met	Response procedure in place.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
144	Install additional transfer and isolation switches.	Reduced number of spurious actuations during a fire.	Yes	B - Intent Met	Items are identified and are being implemented as part of the 805 process. Examples include fuse and alternate feed line modifications to prevent the loss of the 4160 V buses.
145	Enhance fire brigade awareness.	Decreased consequences of a fire.	Yes	B - Intent Met	Most recent inspections and evaluations did not identify any weaknesses in this area.
146	Enhance control of combustibles and ignition sources.	Decreased fire frequency and consequences.	Yes	B - Intent Met	Procedure in place. NFPA-805 project will evaluate the needs for any additional controls.
148	Enhance procedures to mitigate large break LOCA.	Reduced consequences of a large break LOCA.	Yes	B - Intent Met	Existing procedures meet current guidelines issued by the Owner's Group.
149	Install computer aided instrumentation system to assist the operator in assessing post-accident plant status.	Improved prevention of core melt sequences by making operator actions more reliable.	Yes	B - Intent Met	Currently have SPDS in place.
150	Improve maintenance procedures.	Improved prevention of core melt sequences by increasing reliability of important equipment.	Yes	B - Intent Met	Current procedures are in line with industry guidelines and practices.
151	Increase training and operating experience feedback to improve operator response.	Improved likelihood of success of operator actions taken in response to abnormal conditions.	Yes	B - Intent Met	Current training program meets industry standards and practices.
154	Mount or anchor the MCCs to the respective building walls.	Reduces failure probability of MCCs during an earthquake	Yes	B - Intent Met	Identified in the IPEEE and successfully implemented.
155	Install shear pins (or strength bolts) in the AFW pumps.	Takes up the shear load on the pump and/or driver during an earthquake.	Yes	B - Intent Met	Identified in the IPEEE and successfully implemented.
156	Mount all fire extinguishers within their UL Standard required drop height and remove hand-held fire extinguishers from Containment during normal operation.	Reduces the potential for the fire extinguishers to fall during an earthquake and potentially fracturing upon impact with the floor or another object.	Yes	B - Intent Met	Identified in the IPEEE and successfully implemented.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
157	Identify and remove unsecured equipment near areas that contain relays that actuate, so area is kept clear.	Ensures direct access to areas such as Load Shedding and Emergency Load Sequencing (LSELS) and Engineered Safety Feature Actuation System (ESFAS) cabinets. Unsecured equipment (e.g., carts, filing cabinets, and test equipment) in these areas could result	Yes	B - Intent Met	Identified in the IPEEE and successfully implemented.
158	Properly position chain hoists that facilitate maintenance on pumps within pump rooms and institute a training program to ensure that the hoists are properly positioned when not in use.	Improper positioning of hoists reduces the availability due to moving during an earthquake and having chainfalls impacting pump oil bubblers or other soft targets resulting in failure of the pumps.	Yes	B - Intent Met	Identified in the IPEEE and successfully implemented.
159	Secure floor grating to prevent damage to sensing lines due to differential building motion.	Prevent sensing lines that pass through the grating from being damaged.	Yes	B - Intent Met	Identified in the IPEEE and successfully implemented.
166	Installation of high temperature qualified RCP seal O-rings.	Lower potential for RCP seal leakage.	Yes	B - Intent Met	High temperature O-Rings installed.
167	Addition of procedural guidance to re-establish normal service water should essential service water fail.	Provide back-up pumps for UHS cooling.	Yes	B - Intent Met	Procedures in place.
168	Addition of procedural guidance for running charging and safety injection pumps without component cooling water	Allow use of pumps following loss of component cooling water.	Yes	B - Intent Met	Procedures in place.
169	Addition of procedural guidance to verify RHR pump room cooling at switchover to ECCS recirculation phase.	Verifying that support system for RHR pumps is in service to allow continued operation of RHR pumps.	Yes	B - Intent Met	Procedures in place.
170	Modifications to add controls in the main control room to allow remote operation of nearby diesel generator farm and alignment/connection to the plant vital electrical busses.	Faster ability to provide power to the plant electrical busses from the offsite diesel generator farm.	Yes	B - Intent Met	AEPS diesel generators automatically start upon loss of offsite power to the local electrical co-op distribution system. The controls for the breakers to connect to the Callaway distribution system are in the main control room.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
172	Addition of procedural guidance and the required hardware to enable the operators to feed one or more steam generators with a diesel driven firewater pump.	Provide a backup to turbine driven auxiliary feedwater.	Yes	B - Intent Met	Procedure and hardware changes complete
173	Addition of a black start combustion turbine generator.	A redundant source of AC Power that could be used in station blackout events.	Yes	B - Intent Met	The original evaluation of this proposed modification concluded that the cost for this modification was prohibitively high. However, this was subsequently changed and the offsite Alternate Emergency Power System (AEPS) system was installed. The AEPS system consists of diesel generators and a connection to the offsite electrical Co-op.
174	Addition of a black-start engine-generator to provide AC Power during a station blackout	Ability to power a 125VDC battery charger and a charging pump. Powering the battery charger would permit operation of the TDAFP without recovering AC power. Powering a charging pump could provide RCP seal injection and preclude a RCP seal LOCA during a station blackout.	Yes	B - Intent Met	The original evaluation of this proposed modification concluded that the cost for this modification was prohibitively high. However, later implementation of the AEPS system provides the backup power source represented by this item. Also the EC Coordinator Supplemental Guidelines provide procedures and equipment for hookup of a portable generator.
175	Replacement of the positive displacement charging pump with a third centrifugal charging pump.	Provide another source for RCP seal cooling, RCS makeup, and pumped flow for feed and bleed.	Yes	B - Intent Met	The positive displacement charging pump has been replaced by a centrifugal pump that does not require component cooling water. It is powered from a non-safety 4160 VAC power supply.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
176	Provide control modifications to bypass feedwater isolation in order to restore main feedwater.	Allow faster and more reliable bypass of the main feedwater isolation signal in order to restore main feedwater to the steam generators should auxiliary feedwater fail.	Yes	B - Intent Met	Feedwater Isolation bypass switches installed and EOP in place with directions for use.
177	Procedural and hardware modifications to reduce core damage risk due to internal flooding.	The IPE identified a need to form a task force to identify and evaluate potential procedural and hardware modifications aimed at reducing the risk due to internal flooding.	Yes	B - Intent Met	The flooding task force identified 3 generic recommendations; 1) evaluate the impact of the normal charging pump (NCP), 2) evaluate the impact of increased inspections or changes in pipe class on pipe failure probability, and 3) re-analyze pipe break flowrates for actual flow, rather than assuming pump runout flowrates. All three recommendations have been implemented. The flooding analysis credited the NCP and reduced one flood zone below the screening value. A leakage detection program was implemented which uses security personnel and operators to visually inspect specific piping in the major flood zones. The implementation of the leakage detection program reduced flooding risk sufficiently to not require the installation of some watertight doors and piping encapsulation.
140	Increase seismic ruggedness of plant components.	Increased availability of necessary plant equipment during and after seismic events.	Yes	C - Combined	Individual seismic issues identified in the IPEEE are included as SAMA items 154, 155, 156, 157, 158, and 159.

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
141	Provide additional restraints for CO2 tanks.	Increased availability of fire protection given a seismic event.	Yes	C - Combined	Individual seismic issues identified in the IPEEE are included as SAMA items 154, 155, 156, 157, 158, and 159.
1	Provide additional DC battery capacity.	Extended DC power availability during an SBO.	No		Original battery capacity is 4 hrs. No additional battery capacity has been added. Evaluate in Phase II.
2	Replace lead-acid batteries with fuel cells.	Extended DC power availability during an SBO.	No		Plant currently uses batteries rather than fuel cells. Evaluate in Phase II.
5	Provide DC bus cross-ties.	Improved availability of DC power system.	No		No existing capability for DC bus cross-ties. Evaluate in Phase II.
11	Improve 4.16-kV bus cross-tie ability.	Increased availability of on-site AC power.	No		Evaluate during Phase II
15	Install tornado protection on gas turbine generator.	Increased availability of on-site AC power.	No		No gas turbine currently installed. No tornado protection for Alternate Emergency Power System diesel generators. Evaluate in Phase II.
24	Bury off-site power lines.	Improved off-site power reliability during severe weather.	No		Evaluate during Phase II
25	Install an independent active or passive high pressure injection system.	Improved prevention of core melt sequences.	No		Evaluate during Phase II
26	Provide an additional high pressure injection pump with independent diesel.	Reduced frequency of core melt from small LOCA and SBO sequences.	No		Evaluate during Phase II
28	Add a diverse low pressure injection system.	Improved injection capability.	No		Evaluate during Phase II

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
29	Provide capability for alternate injection via diesel-driven fire pump.	Improved injection capability.	No		Currently being evaluated by plant improvement program. Would use unborated water and portable pump (fire truck). Calculation of specific benefit of this SAMA was not performed since it is judged to be potentially low cost. Evaluation will consider impacts of injection of non-borated water.
39	Replace two of the four electric safety injection pumps with diesel-powered pumps.	Reduced common cause failure of the safety injection system. This SAMA was originally intended for the Westinghouse-CE System 80+, which has four trains of safety injection. However, the intent of this SAMA is to provide diversity within the high- and l	No		Evaluate during Phase II
41	Create a reactor coolant depressurization system.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	No		Evaluate during Phase II
43	Add redundant DC control power for SW pumps.	Increased availability of SW.	No		Evaluate during Phase II
46	Add a service water pump.	Increased availability of cooling water.	No		Evaluate during Phase II
54	Increase charging pump lube oil capacity.	Increased time before charging pump failure due to lube oil overheating in loss of cooling water sequences.	No		Evaluate during Phase II
55	Install an independent reactor coolant pump seal injection system, with dedicated diesel.	Reduced frequency of core damage from loss of component cooling water, service water, or station blackout.	No		Evaluate during Phase II
56	Install an independent reactor coolant pump seal injection system, without dedicated diesel.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout.	No		Evaluate during Phase II
58	Install improved reactor coolant pump seals.	Reduced likelihood of reactor coolant pump seal LOCA.	No		Evaluate during Phase II
59	Install an additional component cooling water pump.	Reduced likelihood of loss of component cooling water leading to a reactor coolant pump seal LOCA.	No		Evaluate during Phase II

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie.	Improved ability to cool residual heat removal heat exchangers.	No		Evaluate during Phase II
65	Install a digital feed water upgrade.	Reduced chance of loss of main feed water following a plant trip.	No		Evaluate during Phase II
71	Install a new condensate storage tank (auxiliary feedwater storage tank).	Increased availability of the auxiliary feedwater system.	No		Evaluate during Phase II
77	Provide a passive, secondary-side heat-rejection loop consisting of a condenser and heat sink.	Reduced potential for core damage due to loss-of-feedwater events.	No		Evaluate during Phase II
79	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.	Increased probability of successful feed and bleed.	No		Evaluate during Phase II
80	Provide a redundant train or means of ventilation.	Increased availability of components dependent on room cooling.	No		Evaluate during Phase II
87	Replace service and instrument air compressors with more reliable compressors which have self-contained air cooling by shaft driven fans.	Elimination of instrument air system dependence on service water cooling.	No		Air compressors currently cooled by ESW. Evaluate during Phase II
91	Install a passive containment spray system.	Improved containment spray capability.	No		Evaluate during Phase II
93	Install an unfiltered, hardened containment vent.	Increased decay heat removal capability for non-ATWS events, without scrubbing released fission products.	No		Evaluate during Phase II
94	Install a filtered containment vent to remove decay heat. Option 1: Gravel Bed Filter; Option 2: Multiple Venturi Scrubber	Increased decay heat removal capability for non-ATWS events, with scrubbing of released fission products.	No		Evaluate during Phase II
96	Provide post-accident containment inerting capability.	Reduced likelihood of hydrogen and carbon monoxide gas combustion.	No		Evaluate during Phase II

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
97	Create a large concrete crucible with heat removal potential to contain molten core debris.	Increased cooling and containment of molten core debris. Molten core debris escaping from the vessel is contained within the crucible and a water cooling mechanism cools the molten core in the crucible, preventing melt-through of the base mat.	No		Evaluate during Phase II
98	Create a core melt source reduction system.	Increased cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal from the vitrified compound would be facilitated, and concrete attack would not occur.	No		Evaluate during Phase II
99	Strengthen primary/secondary containment (e.g., add ribbing to containment shell).	Reduced probability of containment over-pressurization.	No		Evaluate during Phase II
100	Increase depth of the concrete base mat or use an alternate concrete material to ensure melt-through does not occur.	Reduced probability of base mat melt-through.	No		Evaluate during Phase II
102	Construct a building to be connected to primary/secondary containment and maintained at a vacuum.	Reduced probability of containment over-pressurization.	No		Evaluate during Phase II
104	Improve leak detection procedures.	Increased piping surveillance to identify leaks prior to complete failure. Improved leak detection would reduce LOCA frequency.	No		Evaluate during Phase II
107	Install a redundant containment spray system.	Increased containment heat removal ability.	No		Evaluate during Phase II

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
108	Install an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel.	Reduced hydrogen detonation potential.	No		Evaluate during Phase II
109	Install a passive hydrogen control system.	Reduced hydrogen detonation potential.	No		Evaluate during Phase II
110	Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure.	Reduced probability of containment failure.	No		Evaluate during Phase II
111	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.	Reduced ISLOCA frequency.	No		Evaluate during Phase II
112	Add redundant and diverse limit switches to each containment isolation valve.	Reduced frequency of containment isolation failure and ISLOCAs.	No		Evaluate during Phase II
113	Increase leak testing of valves in ISLOCA paths.	Reduced ISLOCA frequency.	No		Evaluate during Phase II
114	Install self-actuating containment isolation valves.	Reduced frequency of isolation failure.	No		Evaluate during Phase II
115	Locate residual heat removal (RHR) inside containment	Reduced frequency of ISLOCA outside containment.	No		Evaluate during Phase II
116	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Scrubbed ISLOCA releases.	No		Evaluate during Phase II
119	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage.	Reduced frequency of steam generator tube ruptures.	No		Current frequency of inspection of SG tubes is 100% inspection every third outage. Evaluate during Phase II

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
121	Increase the pressure capacity of the secondary side so that a steam generator tube rupture would not cause the relief valves to lift.	Eliminates release pathway to the environment following a steam generator tube rupture.	No		Evaluate during Phase II
122	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture	Enhanced depressurization capabilities during steam generator tube rupture.	No		Evaluate during Phase II
125	Route the discharge from the main steam safety valves through a structure where a water spray would condense the steam and remove most of the fission products.	Reduced consequences of a steam generator tube rupture.	No		Evaluate during Phase II
126	Install a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources	Reduced consequences of a steam generator tube rupture.	No		Evaluate during Phase II
129	Vent main steam safety valves in containment.	Reduced consequences of a steam generator tube rupture.	No		Evaluate during Phase II
130	Add an independent boron injection system.	Improved availability of boron injection during ATWS.	No		Evaluate during Phase II
131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Improved equipment availability after an ATWS.	No		Evaluate during Phase II
133	Install an ATWS sized filtered containment vent to remove decay heat.	Increased ability to remove reactor heat from ATWS events.	No		Evaluate during Phase II
136	Install motor generator set trip breakers in control room.	Reduced frequency of core damage due to an ATWS.	No		Evaluate during Phase II
147	Install digital large break LOCA protection system.	Reduced probability of a large break LOCA (a leak before break).	No		Evaluate during Phase II
153	Install secondary side guard pipes up to the main steam isolation valves.	Prevents secondary side depressurization should a steam line break occur upstream of the main steam isolation valves. Also guards against or prevents consequential multiple steam generator tube ruptures following a main steam line break event.	No		Evaluate during Phase II

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
160	Modifications to lessen impact of internal flooding path through Control Building dumbwaiter.	Lower impact of flood that propagates through the dumbwaiter	No		Evaluate during Phase II
161	Improvements to PORV performance that will lower the probability of failure to open.	Decrease in risk due to PORV failing to open.	No		Evaluate during Phase II
162	Install a large volume EDG fuel oil tank at an elevation greater than the EDG fuel oil day tanks.	Allows transfer of EDF fuel oil to the EDG day tanks on failure of the fuel oil transfer pumps.	No		Evaluate during Phase II
163	Improve feedwater check valve reliability to reduce probability of failure to open.	Lower risk due to failures in which feedwater check valves fail to open and allow feeding of the steam generators.	No		Valves replaced with new type, but are still significant risk contributor. Evaluate in Phase II.
164	Provide the capability to power the normal service water pumps from AEPS.	Provide backup to ESW in conditions with power only available from AEPS.	No		Evaluate during Phase II
171	Increase the size of the RWST or otherwise improve the availability of the RWST	Ensure a supply of makeup water is available from the RWST.	No		Evaluate during Phase II
178	Improvements to UHS cooling tower electrical room HVAC.	Improve availability or mitigate loss of HVAC.	No		Evaluate during Phase II
179	Modify procedures such that the water loop seals in the RCS cold legs are not cleared following core damage.	Prevents possible thermally induced steam generator tube rupture following core damage.	No		Evaluate during Phase II
180	Install lower amperage fuses for various 14 AWG control circuits in the MCR. The majority of the modification centers around the trip circuit fuses on NB, NG, PA, PB, and PG system breakers.	Reduced fire risk.	No		Evaluate during Phase II
181	Install redundant fuses and isolation switches for MCR evacuation procedure OTO-ZZ-00001.	Reduced fire risk.	No		Evaluate during Phase II

Table 6-1. Callaway Plant Phase I SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	Screened Out Ph 1?	Screening Criterion	Phase I Disposition
182	To protect against multiple spurious operation scenarios, cable runs will be changed to run a single wire in a protected metal jacket such that spurious valve opening due to a hot short affecting the valve control circuit is eliminated for the fire area. This modification will be implemented in multiple fire areas.	Reduced fire risk.	No		Evaluate during Phase II
183	Quick response sprinkler heads in cable chases A-11, C-30, and C-31 will be modified to be in accordance with the applicable requirements of NFPA 13-1976 edition.	Reduced fire risk.	No		Evaluate during Phase II
184	Improvements in the reliability of the Steam Line Isolation automatic signal.	More reliable main steam line isolation.	No		Evaluate during Phase II
185	Automate initiation of CCW flow to the RHR heat exchangers.	More reliable than manual initiation of flow to RRHR HX.	No		Evaluate during Phase II
186	Develop a procedure and obtain equipment to provide a temporary hookup of fire water as a replacement for ESW	Backup cooling water if ESW/SW is lost	No		Evaluate during Phase II

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
1	Provide additional DC battery capacity.	Extended DC power availability during an SBO.	12.17%	10.87%	NOSBO	No Station Blackout Events	$360K	>$1M	10.49%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
2	Replace lead-acid batteries with fuel cells.	Extended DC power availability during an SBO.	12.17%	10.87%	NOSBO	No Station Blackout Events	$360K	>$1M	10.49%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
5	Provide DC bus cross-ties.	Improved availability of DC power system.	0.30%	0.00%	DC01	TDAFW no DC Dependency	$1K	>$199K	0.03%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
11	Improve 4.16-kV bus cross-tie ability.	Increased availability of on-site AC power.	12.17%	10.87%	NOSBO	No Station Blackout Events	$360K	>$1M	10.49%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Cost for implementation includes analysis, material to be purchased and prestaged, development of procedures, and training of personnel on implementation.,
15	Install tornado protection on gas turbine generator.	Increased availability of on-site AC power.	2.65%	4.35%	LOSP1	No tornado related LOSP	$91K	>$500K	3.38%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
24	Bury off-site power lines.	Improved off-site power reliability during severe weather.	40.66%	41.30%	NOLOSP	Eliminate all Loss of Offsite Power Events	$1.2M	>$3M	35.28%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Previous SAMA submittals have estimated approximately $1M per mile.
25	Install an independent active or passive high pressure injection system.	Improved prevention of core melt sequences.	2.77%	0.00%	LOCA12	No failures of the charging or SI pumps	$48K	>$1M	0.35%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
26	Provide an additional high pressure injection pump with independent diesel.	Reduced frequency of core melt from small LOCA and SBO sequences.	2.77%	0.00%	LOCA12	No failures of the charging or SI pumps	$48K	>$1M	0.35%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
28	Add a diverse low pressure injection system.	Improved injection capability.	3.19%	2.17%	LOCA03	No failure of low pressure injection	$65K	>$1M	1.01%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
29	Provide capability for alternate injection via diesel-driven fire pump.	Improved injection capability.									Potentially Cost-Beneficial	SAMA is judged to be low cost, but analysis is needed to determine impacts of injection of non-borated water to RCS. Expert Panel judged this SAMA to be potentially cost-beneficial without determining an actual benefit or cost.

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
39	Replace two of the four electric safety injection pumps with diesel-powered pumps.	Reduced common cause failure of the safety injection system. This SAMA was originally intended for the Westinghouse-CE System 80+, which has four trains of safety injection. However, the intent of this SAMA is to provide diversity within the high- and l	2.77%	0.00%	LOCA12	No failures of the charging or SI pumps	$48K	>$1M	0.35%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
41	Create a reactor coolant depressurization system.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	0.78%	0.00%	DEPRESS	No failures of depressurization	$12K	>$500K	0.27%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
43	Add redundant DC control power for SW pumps.	Increased availability of SW.	0.30%	0.00%	SW01	Service Water Pumps not dependent on DC Power	$1K	>$100K	0.06%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
46	Add a service water pump.	Increased availability of cooling water.	17.60%	27.72%	SW02	No failures of ESW pumps	$636K	>$5M	23.26%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
54	Increase charging pump lube oil capacity.	Increased time before charging pump failure due to lube oil overheating in loss of cooling water sequences.	0.48%	0.00%	CHG01	Charging pumps not dependent on cooling water.	$4K	>$100K	0.06%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
55	Install an independent reactor coolant pump seal injection system, with dedicated diesel.	Reduced frequency of core damage from loss of component cooling water, service water, or station blackout.	5.54%	0.00%	RCPLOCA	No RCP Seal LOCAs	$94K	>$1M	0.21%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Previous investigation into installing such a system concluded that operators did not have sufficient time to place the system in service prior to seal damage.
56	Install an independent reactor coolant pump seal injection system, without dedicated diesel.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout.	5.54%	0.00%	RCPLOCA	No RCP Seal LOCAs	$94K	>$500K	0.21%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
58	Install improved reactor coolant pump seals.	Reduced likelihood of reactor coolant pump seal LOCA.	5.54%	0.00%	RCPLOCA	No RCP Seal LOCAs	$94K	>$3M	0.21%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
59	Install an additional component cooling water pump.	Reduced likelihood of loss of component cooling water leading to a reactor coolant pump seal LOCA.	3.61%	0.00%	CCW01	No failures of the CCW Pumps	$59K	>$1M	0.07%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie.	Improved ability to cool residual heat removal heat exchangers.	5.39%	0.76%	FWCCW 2	Evaluate fire water hookup to RHR HX	$104K	<150K	0.77%%	Expert Panel	Potentially Cost Beneficial	The cost estimate is for development of a procedure and use of temporary connections. Cost of permanent modification would be significantly higher.
65	Install a digital feed water upgrade.	Reduced chance of loss of main feed water following a plant trip.	1.57%	0.00%	FW01	No loss of Feedwater Events	$29K	$19M	0.49%	Callaway Modification Costs	Not Cost-Beneficial	Cost will exceed benefit.
71	Install a new condensate storage tank (auxiliary feedwater storage tank).	Increased availability of the auxiliary feedwater system.	1.14%	0.00%	CST01	CST does not deplete	$18K	>$2.5M	0.24%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
77	Provide a passive, secondary-side heat-rejection loop consisting of a condenser and heat sink.	Reduced potential for core damage due to loss-of-feedwater events.	1.57%	0.00%	FW01	No loss of Feedwater Events	$29K	$>1M	0.49%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
79	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.	Increased probability of successful feed and bleed.	3.43%	2.17%	FB01	Only one PORV required for Feed & Bleed	$79K	>$500K	1.68%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
80	Provide a redundant train or means of ventilation.	Increased availability of components dependent on room cooling.	6.08%	4.35%	HVAC	No dependencies on HVAC	$156K	>$1M	3.87%	Expert Panel	Potentially Cost Beneficial	Procedures to open doors or provide temporary ventilation may be cost beneficial for the EDGs, MDAFW pumps, and charging pumps. Procedures for opening doors to the DC switchgear rooms exist.
87	Replace service and instrument air compressors with more reliable compressors which have self-contained air cooling by shaft driven fans.	Elimination of instrument air system dependence on service water cooling.	0.36%	0.00%	INSTAIR	Eliminate all instrument air failures	$2K	>$500K	0.06%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
91	Install a passive containment spray system.	Improved containment spray capability.	19.52%	36.96%	CONT01	No failures due to containment overpressure	$793K	>$10M	31.32%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
93	Install an unfiltered, hardened containment vent.	Increased decay heat removal capability for non-ATWS events, without scrubbing released fission products.	19.52%	36.96%	CONT01	No failures due to containment overpressure	$793K	>$2M	31.32%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
94	Install a filtered containment vent to remove decay heat. Option 1: Gravel Bed Filter; Option 2: Multiple Venturi Scrubber	Increased decay heat removal capability for non-ATWS events, with scrubbing of released fission products.	19.52%	36.96%	CONT01	No failures due to containment overpressure	$793K	>$2M	31.32%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
96	Provide post-accident containment inerting capability.	Reduced likelihood of hydrogen and carbon monoxide gas combustion.	0.48%	0.00%	H2BURN	No hydrogen burns/explosions	$10K	>$100K	0.44%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
97	Create a large concrete crucible with heat removal potential to contain molten core debris.	Increased cooling and containment of molten core debris. Molten core debris escaping from the vessel is contained within the crucible and a water cooling mechanism cools the molten core in the crucible, preventing melt-through of the base mat.			MAB			>$10M	Note 1	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
98	Create a core melt source reduction system.	Increased cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal from the vitrified compound would be facilitated, and concrete attack would not occur.			MAB			>$10M	Note 1	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
99	Strengthen primary/secondary containment (e.g., add ribbing to containment shell).	Reduced probability of containment over-pressurization.	19.52%	36.96%	CONT01	No failures due to containment overpressure	$1.2M	>$10M	31.32%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
100	Increase depth of the concrete base mat or use an alternate concrete material to ensure melt-through does not occur.	Reduced probability of base mat melt-through.			MAB			>$10M	Note 1	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
102	Construct a building to be connected to primary/secondary containment and maintained at a vacuum.	Reduced probability of containment over-pressurization.	19.52%	36.96%	CONT01	No failures due to containment overpressure	$1.2M	>$10M	31.32%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
104	Improve leak detection procedures.	Increased piping surveillance to identify leaks prior to complete failure. Improved leak detection would reduce LOCA frequency.	39.34%	2.17%	LOCA05	No piping system LOCAs	$689K	>$2M	1.03%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
107	Install a redundant containment spray system.	Increased containment heat removal ability.	19.52%	36.96%	CONT01	No failures due to containment overpressure	$1.2M	>$2M	31.32%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
108	Install an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel.	Reduced hydrogen detonation potential.	0.48%	0.00%	H2BURN	No hydrogen burns/explosions	$10K	>$100K	0.44%	Expert Panel	Not Cost-Beneficial	
109	Install a passive hydrogen control system.	Reduced hydrogen detonation potential.	0.48%	0.00%	H2BURN	No hydrogen burns/explosions	$10K	>$100M	0.44%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
110	Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure.	Reduced probability of containment failure.			MAB			>$10M	Note 1	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
111	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.	Reduced ISLOCA frequency.	1.33%	8.70%	ISLOCA	No ISLOCA events	$123K	>$500K	7.08%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
112	Add redundant and diverse limit switches to each containment isolation valve.	Reduced frequency of containment isolation failure and ISLOCAs.	0.30%	0.00%	CONT02	No failures of containment isolation	$1K	>$1M		Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
113	Increase leak testing of valves in ISLOCA paths.	Reduced ISLOCA frequency.	1.33%	8.70%	ISLOCA	No ISLOCA events	$123K	>$1M	7.08%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
114	Install self-actuating containment isolation valves.	Reduced frequency of isolation failure.	0.30%	0.00%	CONT02	No failures of containment isolation	$1K	>$500K	0.03%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
115	Locate residual heat removal (RHR) inside containment	Reduced frequency of ISLOCA outside containment.	1.33%	8.70%	ISLOCA	No ISLOCA events	$123K	>$1M	7.08%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
116	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Scrubbed ISLOCA releases.	1.33%	8.70%	ISLOCA	No ISLOCA events	$123K	>$1M	7.08%	Expert Panel	Not Cost-Beneficial	Cost would exceed benefit. Current plant design requires drains to be open. Analysis and license changes required to implement are included in the cost estimate.
119	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage.	Reduced frequency of steam generator tube ruptures.	20.47%	63.28%	NOSGTR	No SGTR Events	$1.4M	>$3M	69.43%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
121	Increase the pressure capacity of the secondary side so that a steam generator tube rupture would not cause the relief valves to lift.	Eliminates release pathway to the environment following a steam generator tube rupture.	20.47%	63.28%	NOSGTR	No SGTR Events	$1.4M	>$10M	69.43%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
122	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture	Enhanced depressurization capabilities during steam generator tube rupture.	20.47%	63.28%	NOSGTR	No SGTR Events	$1.4M	>$10M	69.43%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
125	Route the discharge from the main steam safety valves through a structure where a water spray would condense the steam and remove most of the fission products.	Reduced consequences of a steam generator tube rupture.	20.47%	63.28%	NOSGTR	No SGTR Events	$1.4M	>$10M	69.43%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
126	Install a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources	Reduced consequences of a steam generator tube rupture.	20.47%	63.28%	NOSGTR	No SGTR Events	$1.4M	>$10M	69.43%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
129	Vent main steam safety valves in containment.	Reduced consequences of a steam generator tube rupture.	20.47%	63.28%	NOSGTR	No SGTR Events	$1.4M	>$10M	69.43%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Current containment design does not support this modification. Modifications to containment and associated analysis are included in the cost estimate.
130	Add an independent boron injection system.	Improved availability of boron injection during ATWS.	2.41%	2.17%	NOATWS	Eliminate all ATWS	$63K	>$1M	1.85%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Improved equipment availability after an ATWS.	2.41%	2.17%	NOATWS	Eliminate all ATWS	$63K	>$2M	1.85%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
133	Install an ATWS sized filtered containment vent to remove decay heat.	Increased ability to remove reactor heat from ATWS events.	2.41%	2.17%	NOATWS	Eliminate all ATWS	$63K	>$1M	1.85%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit
136	Install motor generator set trip breakers in control room.	Reduced frequency of core damage due to an ATWS.	2.41%	2.17%	NOATWS	Eliminate all ATWS	$63K	>$500K	1.85%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
147	Install digital large break LOCA protection system.	Reduced probability of a large break LOCA (a leak before break).	39.34%	2.17%	LOCA05	No piping system LOCAs	$689K	>$5M	1.03%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
153	Install secondary side guard pipes up to the main steam isolation valves.	Prevents secondary side depressurization should a steam line break occur upstream of the main steam isolation valves. Also guards against or prevents consequential multiple steam generator tube ruptures following a main steam line break event.	2.53%	0.00%	NOSLB	No Steam Line Breaks	$51K	>$1M	0.87%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
160	Modifications to lessen impact of internal flooding path through Control Building dumbwaiter.	Lower impact of flood that propagates through the dumbwaiter						<$50K		Expert Panel	Potentially Cost-Beneficial	Relatively minor modifications to door opening could result in lower flow to the dumbwaiter. Specific benefit could not be calculated but SAMA item is judged to be low cost and therefore potentially cost beneficial.
161	Improvements to PORV performance that will lower the probability of failure to open.	Decrease in risk due to PORV failing to open.	0.85%	0.46%	PORV	PORVs do not fail to open	$18K	>$100K	0.24%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
162	Install a large volume EDG fuel oil tank at an elevation greater than the EDG fuel oil day tanks.	Allows transfer of EDF fuel oil to the EDG day tanks on failure of the fuel oil transfer pumps.	1.14%	7.60%	EDGFUEL	No EDG fuel pump failures	$124K	$150K	7.11%	Wolf Creek	Potentially Cost-Beneficial	Wolf Creek estimated cost of $150K is less than the potential benefit.
163	Improve feedwater check valve reliability to reduce probability of failure to open.	Lower risk due to failures in which feedwater check valves fail to open and allow feeding of the steam generators.	5.52%	2.05%	FW02	Feedwater Check Valves do not fail to open	$127K	>$500K	2.23%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
164	Provide the capability to power the normal service water pumps from AEPS.	Provide backup to ESW in conditions with power only available from AEPS.	5.62%	7.64%	SW03	AEPS power to SW pumps	$191K	>$500K	6.37%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
171	Increase the size of the RWST or otherwise improve the availability of the RWST	Ensure a supply of makeup water is available from the RWST.	0.68%	0.13%	LOCA04	RWST does not deplete	$13K	>$100K	0.07%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
178	Improvements to UHS cooling tower electrical room HVAC.	Improve availability or mitigate loss of HVAC.	3.29%	4.75% %	HVAC02	UHS cooling tower electrical room HVAC does not fail.	$113K		3.82%	Expert Panel	Potentially Cost Beneficial	Implementation of temporary ventilation or opening of doors will be a lower cost than the calculated benefit.
179	Modify procedures such that the water loop seals in the RCS cold legs are not cleared following core damage.	Prevents possible thermally induced steam generator tube rupture following core damage.	0.15%	3.18%	RAI7a	Reduced probability of thermally induced steam generator tube failure	$63K		4.46%	Expert Panel	Potentially Cost Beneficial	Implementation of procedure change will be lower cost than benefit, especially if 95% CDF benefit is considered.
180	Install lower amperage fuses for various 14 AWG control circuits in the MCR. The majority of the modification centers around the trip circuit fuses on NB, NG, PA, PB, and PG system breakers.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	% Red. In CDF	% Red. In OS Dose	SAMA Case	SAMA Case Description	Benefit	Cost	% Red IN OECR	Cost Basis	Evaluation	Basis for Evaluation
181	Install redundant fuses and isolation switches for MCR evacuation procedure OTO-ZZ-00001.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
182	To protect against multiple spurious operation scenarios, cable runs will be changed to run a single wire in a protected metal jacket such that spurious valve opening due to a hot short affecting the valve control circuit is eliminated for the fire area. This modification will be implemented in multiple fire areas.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
183	Quick response sprinkler heads in cable chases A-11, C-30, and C-31 will be modified to be in accordance with the applicable requirements of NFPA 13-1976 edition.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
184	Improvements in the reliability of the Steam Line Isolation automatic signal.	More reliable main steam line isolation.	0.59%	0.95%	SLIS	Steam Line Isolation System does not fail	$28K	>500K	1.06%	Expert Panel	Not Cost-Beneficial	Cost is for installation of redundant instrumentation system and would likely be much higher. Procedure and training already direct operators to manually back up failed automatic actuations.
185	Automate initiation of CCW flow to the RHR heat exchangers.	More reliable than manual initiation of flow to RRHR HX.	3.53%	0.14%	HEP	Evaluate automating CCW flow to RHR HXs	$62K	>$500K	0.11%	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
186	Develop a procedure and obtain equipment to provide a temporary hookup of fire water as a replacement for ESW	Backup cooling water if ESW/SW is lost	17.60%	27.72%	SW02	No failures of ESW pumps	$636K	>$1M	23.26%	Expert Panel	Potentially Cost Beneficial	Ability to do this will require larger fire pumps

Table 7-1. Callaway Plant 1 Phase II SAMA Analysis (Continued)

	OS = off site Note 1: For SAMA items that were judged to cost significantly more than the Maximum Attainable Benefit (MAB), no calculation of the individual benefit was performed.

Table 8-1. Callaway Plant Sensitivity Evaluation

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
1	Provide additional DC battery capacity.	Extended DC power availability during an SBO.	NOSBO	$360K	$588K	$325K	$512K	$761K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
2	Replace lead-acid batteries with fuel cells.	Extended DC power availability during an SBO.	NOSBO	$360K	$588K	$325K	$512K	$761K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
5	Provide DC bus cross-ties.	Improved availability of DC power system.	DC01	$1K	$1K	$1K	$1K	$1K	>$199K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
11	Improve 4.16-kV bus cross-tie ability.	Increased availability of on-site AC power.	NOSBO	$360K	$588K	$325K	$512K	$761K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Cost for implementation includes analysis, material to be purchased and prestaged, development of procedures, and training of personnel on implementation.,
15	Install tornado protection on gas turbine generator.	Increased availability of on-site AC power.	LOSP1	$91K	$144K	$82K	$125K	$192K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
24	Bury off-site power lines.	Improved off-site power reliability during severe weather.	NOLOSP	$1.2M	$2.0M	$1.1M	$1.7M	$2.6M	>$3M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Previous SAMA submittals have estimated approximately $1M per mile.
25	Install an independent active or passive high pressure injection system.	Improved prevention of core melt sequences.	LOCA12	$48K	$85K	$44K	$75	$102	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
26	Provide an additional high pressure injection pump with independent diesel.	Reduced frequency of core melt from small LOCA and SBO sequences.	LOCA12	$48K	$85K	$44K	$75	$102	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
28	Add a diverse low pressure injection system.	Improved injection capability.	LOCA03	$65K	$111K	$58K	$97K	$137K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
29	Provide capability for alternate injection via diesel-driven fire pump.	Improved injection capability.									Potentially Cost-Beneficial	SAMA is judged to be low cost, but analysis is needed to determine impacts of injection of non-borated water to RCS. Expert Panel judged this SAMA to be potentially cost-beneficial without determining an actual benefit or cost.

Table 8-1. Callaway Plant Sensitivity Evaluation (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
39	Replace two of the four electric safety injection pumps with diesel-powered pumps.	Reduced common cause failure of the safety injection system. This SAMA was originally intended for the Westinghouse-CE System 80+, which has four trains of safety injection. However, the intent of this SAMA is to provide diversity within the high- and l	LOCA12	$48K	$85K	$44K	$75	$102	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
41	Create a reactor coolant depressurization system.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	DEPRESS	$12K	$20K	$11K	$17K	$25K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
43	Add redundant DC control power for SW pumps.	Increased availability of SW.	SW01	$1K	$2K	$1K	$2K	$3K	>$100K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
46	Add a service water pump.	Increased availability of cooling water.	SW02	$636K	$1M	$575K	$879K	$1.3M	>$5M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
54	Increase charging pump lube oil capacity.	Increased time before charging pump failure due to lube oil overheating in loss of cooling water sequences.	CHG01	$4K	$7K	$4K	$6K	$9K	>$100K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
55	Install an independent reactor coolant pump seal injection system, with dedicated diesel.	Reduced frequency of core damage from loss of component cooling water, service water, or station blackout.	RCPLOCA	$94K	$168K	$85K	$148K	$198K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Previous investigation into installing such a system concluded that operators did not have sufficient time to place the system in service prior to seal damage.
56	Install an independent reactor coolant pump seal injection system, without dedicated diesel.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout.	RCPLOCA	$94K	$168K	$85K	$148K	$198K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
58	Install improved reactor coolant pump seals.	Reduced likelihood of reactor coolant pump seal LOCA.	RCPLOCA	$94K	$168K	$85K	$148K	$198K	>$3M		Not Cost-Beneficial	Cost will exceed benefit.
59	Install an additional component cooling water pump.	Reduced likelihood of loss of component cooling water leading to a reactor coolant pump seal LOCA.	CCW01	$59K	$106K	$53K	$93K	$124K	>$1M	Cost will exceed benefit	Not Cost-Beneficial	Cost will exceed benefit.

Table 8-1. Callaway Plant Sensitivity Evaluation (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie.	Improved ability to cool residual heat removal heat exchangers.	FWCCW2	$104K	$184K	$94K	$161K	$220K	<150K	Expert Panel	Potentially Cost Beneficial	The cost estimate is for development of a procedure and use of temporary connections. Cost of permanent modification would be significantly higher.
65	Install a digital feed water upgrade.	Reduced chance of loss of main feed water following a plant trip.	FW01	$29K	$50K	$27K	$44K	$62K	$19M	Callaway Modification Costs	Not Cost-Beneficial	Cost will exceed benefit.
71	Install a new condensate storage tank (auxiliary feedwater storage tank).	Increased availability of the auxiliary feedwater system.	CST01	$18K	$32K	$16K	$28K	$39K	>$2.5M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
77	Provide a passive, secondary-side heat-rejection loop consisting of a condenser and heat sink.	Reduced potential for core damage due to loss-of-feedwater events.	FW01	$29K	$50K	$27K	$44K	$62K	$>1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
79	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.	Increased probability of successful feed and bleed.	FB01	$79K	$133K	$72K	$117K	$168K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
80	Provide a redundant train or means of ventilation.	Increased availability of components dependent on room cooling.	HVAC	$156K	$259K	$141K	$227K	$331K	>$1M	Expert Panel	Potentially Cost Beneficial	Procedures to open doors or provide temporary ventilation may be cost beneficial for the EDGs, MDAFW pumps, and charging pumps. Procedures for opening doors to the DC switchgear rooms exist.
87	Replace service and instrument air compressors with more reliable compressors which have self-contained air cooling by shaft driven fans.	Elimination of instrument air system dependence on service water cooling.	INSTAIR	$2K	$3K	$2K	$$2K	$4K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
91	Install a passive containment spray system.	Improved containment spray capability.	CONT01	$793K	$1.2M	$717K	$1.1M	$1.7M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
93	Install an unfiltered, hardened containment vent.	Increased decay heat removal capability for non-ATWS events, without scrubbing released fission products.	CONT01	$793K	$1.2M	$717K	$1.1M	$1.7M	>$2M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
94	Install a filtered containment vent to remove decay heat. Option 1: Gravel Bed Filter; Option 2: Multiple Venturi Scrubber	Increased decay heat removal capability for non-ATWS events, with scrubbing of released fission products.	CONT01	$793K	$1.2M	$717K	$1.1M	$1.7M	>$2M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
96	Provide post-accident containment inerting capability.	Reduced likelihood of hydrogen and carbon monoxide gas combustion.	H2BURN	$10K	$15K	$9K	$13K	$20K	>$100K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 8-1. Callaway Plant Sensitivity Evaluation (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
97	Create a large concrete crucible with heat removal potential to contain molten core debris.	Increased cooling and containment of molten core debris. Molten core debris escaping from the vessel is contained within the crucible and a water cooling mechanism cools the molten core in the crucible, preventing melt-through of the base mat.	MAB						>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
98	Create a core melt source reduction system.	Increased cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal from the vitrified compound would be facilitated, and concrete attack would not occur.	MAB						>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
99	Strengthen primary/secondary containment (e.g., add ribbing to containment shell).	Reduced probability of containment over-pressurization.	CONT01	$1.2M	$1.2M	$717K	$1.1M	$1.7M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
100	Increase depth of the concrete base mat or use an alternate concrete material to ensure melt-through does not occur.	Reduced probability of base mat melt-through.	MAB						>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
102	Construct a building to be connected to primary/secondary containment and maintained at a vacuum.	Reduced probability of containment over-pressurization.	CONT01	$1.2M	$1.2M	$717K	$1.1M	$1.7M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
104	Improve leak detection procedures.	Increased piping surveillance to identify leaks prior to complete failure. Improved leak detection would reduce LOCA frequency.	LOCA05	$685K	$1.2M	$620K	$1.1M	$1.5M	>$2M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
107	Install a redundant containment spray system.	Increased containment heat removal ability.	CONT01	$1.2M	$1.2M	$717K	$1.1M	$1.7M	>$2M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
108	Install an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel.	Reduced hydrogen detonation potential.	H2BURN	$10K	$15K	$9K	$13K	$20K	>$100K	Expert Panel	Not Cost-Beneficial	
109	Install a passive hydrogen control system.	Reduced hydrogen detonation potential.	H2BURN	$10K	$15K	$9K	$13K	$20K	>$100M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
110	Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure.	Reduced probability of containment failure.	MAB						>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 8-1. Callaway Plant Sensitivity Evaluation (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
111	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.	Reduced ISLOCA frequency.	ISLOCA	$123K	$179K	$111K	$154K	$259K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
112	Add redundant and diverse limit switches to each containment isolation valve.	Reduced frequency of containment isolation failure and ISLOCAs.	CONT02	$1K	$1K	$1K	$1K	$2K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
113	Increase leak testing of valves in ISLOCA paths.	Reduced ISLOCA frequency.	ISLOCA	$123K	$179K	$111K	$154K	$259K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
114	Install self-actuating containment isolation valves.	Reduced frequency of isolation failure.	CONT02	$1K	$1K	$1K	$1K	$2K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
115	Locate residual heat removal (RHR) inside containment	Reduced frequency of ISLOCA outside containment.	ISLOCA	$123K	$179K	$111K	$154K	$259K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
116	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Scrubbed ISLOCA releases.	ISLOCA	$123K	$179K	$111K	$154K	$259K	>$1M	Expert Panel	Not Cost-Beneficial	Cost would exceed benefit. Current plant design requires drains to be open. Analysis and license changes required to implement are included in the cost estimate.
119	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage.	Reduced frequency of steam generator tube ruptures.	NOSGTR	$1.4M	$2.1M	$1.2M	$1.8M	$2.9M	>$3M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
121	Increase the pressure capacity of the secondary side so that a steam generator tube rupture would not cause the relief valves to lift.	Eliminates release pathway to the environment following a steam generator tube rupture.	NOSGTR	$1.4M	$2.1M	$1.2M	$1.8M	$2.9M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
122	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture	Enhanced depressurization capabilities during steam generator tube rupture.	NOSGTR	$1.4M	$2.1M	$1.2M	$1.8M	$2.9M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
125	Route the discharge from the main steam safety valves through a structure where a water spray would condense the steam and remove most of the fission products.	Reduced consequences of a steam generator tube rupture.	NOSGTR	$1.4M	$2.1M	$1.2M	$1.8M	$2.9M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
126	Install a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources	Reduced consequences of a steam generator tube rupture.	NOSGTR	$1.4M	$2.1M	$1.2M	$1.8M	$2.9M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
129	Vent main steam safety valves in containment.	Reduced consequences of a steam generator tube rupture.	NOSGTR	$1.4M	$2.1M	$1.2M	$1.8M	$2.9M	>$10M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit. Current containment design does not support this modification. Modifications to containment and associated analysis are included in the cost estimate.

Table 8-1. Callaway Plant Sensitivity Evaluation (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
130	Add an independent boron injection system.	Improved availability of boron injection during ATWS.	NOATWS	$63K	$104K	$57K	$90K	$134K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Improved equipment availability after an ATWS.	NOATWS	$63K	$104K	$57K	$90K	$134K	>$2M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
133	Install an ATWS sized filtered containment vent to remove decay heat.	Increased ability to remove reactor heat from ATWS events.	NOATWS	$63K	$104K	$57K	$90K	$134K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit
136	Install motor generator set trip breakers in control room.	Reduced frequency of core damage due to an ATWS.	NOATWS	$63K	$104K	$57K	$90K	$134K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
147	Install digital large break LOCA protection system.	Reduced probability of a large break LOCA (a leak before break).	LOCA05	$689K	$1.2M	$620K	$1.1M	$1.5M	>$5M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
153	Install secondary side guard pipes up to the main steam isolation valves.	Prevents secondary side depressurization should a steam line break occur upstream of the main steam isolation valves. Also guards against or prevents consequential multiple steam generator tube ruptures following a main steam line break event.	NOSLB	$51K	$87K	$46K	$77K	$108K	>$1M	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
160	Modifications to lessen impact of internal flooding path through Control Building dumbwaiter.	Lower impact of flood that propagates through the dumbwaiter							<$50K	Expert Panel	Potentially Cost-Beneficial	Relatively minor modifications to door opening could result in lower flow to the dumbwaiter. Specific benefit could not be calculated but SAMA item is judged to be low cost and therefore potentially cost beneficial.
161	Improvements to PORV performance that will lower the probability of failure to open.	Decrease in risk due to PORV failing to open.	PORV	$18K	$32K	$16K	$28K	$39K	>$100K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
162	Install a large volume EDG fuel oil tank at an elevation greater than the EDG fuel oil day tanks.	Allows transfer of EDF fuel oil to the EDG day tanks on failure of the fuel oil transfer pumps.	EDGFUEL	$124K	$131K	$113K	$156K	$263K	$150K	Wolf Creek	Potentially Cost-Beneficial	Wolf Creek estimated cost of $150K is less than the potential benefit.
163	Improve feedwater check valve reliability to reduce probability of failure to open.	Lower risk due to failures in which feedwater check valves fail to open and allow feeding of the steam generators.	FW02	$127K	$218K	$115K	$191K	$270K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
164	Provide the capability to power the normal service water pumps from AEPS.	Provide backup to ESW in conditions with power only available from AEPS.	SW03	$1191K	$307K	$172K	$267K	$403K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
171	Increase the size of the RWST or otherwise improve the availability of the RWST	Ensure a supply of makeup water is available from the RWST.	LOCA04	$13K	$23K	$12K	$20K	$27K	>$100K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.

Table 8-1. Callaway Plant Sensitivity Evaluation (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
178	Improvements to UHS cooling tower electrical room HVAC.	Improve availability or mitigate loss of HVAC.	HVAC02	$113K	$181K	$102K	$158K	$239K	<100K	Expert Panel	Potentially Cost Beneficial	Implementation of temporary ventilation or opening of doors will be a lower cost than the calculated benefit.
179	Modify procedures such that the water loop seals in the RCS cold legs are not cleared following core damage.	Prevents possible thermally induced steam generator tube rupture following core damage.	RAI7a	$63K	$87K	$57K	$75K	$134K		Expert Panel	Potentially Cost Beneficial	Implementation of procedure change will be lower cost than benefit, especially if 95% CDF benefit is considered.
180	Install lower amperage fuses for various 14 AWG control circuits in the MCR. The majority of the modification centers around the trip circuit fuses on NB, NG, PA, PB, and PG system breakers.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amandment request committed to performing the modification.
181	Install redundant fuses and isolation switches for MCR evacuation procedure OTO-ZZ-00001.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
182	To protect against multiple spurious operation scenarios, cable runs will be changed to run a single wire in a protected metal jacket such that spurious valve opening due to a hot short affecting the valve control circuit is eliminated for the fire area. This modification will be implemented in multiple fire areas.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
183	Quick response sprinkler heads in cable chases A-11, C-30, and C-31 will be modified to be in accordance with the applicable requirements of NFPA 13-1976 edition.	Reduced fire risk.									Potentially Cost Beneficial	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.

Table 8-1. Callaway Plant Sensitivity Evaluation (Continued)

Callaway SAMA Number	Potential Improvement	Discussion	SAMA Case	Benefit	Benefit at 3% Disc Rate	Benefit at Realistic Disc Rate	Benefit at 33yrs	Benefit at 95% CDF	Cost	Cost Basis	Evaluation	Basis for Evaluation
184	Improvements in the reliability of the Steam Line Isolation automatic signal.	More reliable main steam line isolation.	SLIS	$28K	$40K	$23K	$35K	$55K	>$500K	Expert Panel	Not Cost-Beneficial	Cost is for installation of redundant instrumentation system and would likely be much higher. Procedure and training already direct operators to manually back up failed automatic actuations.
185	Automate initiation of CCW flow to the RHR heat exchangers.	More reliable than manual initiation of flow to RRHR HX.	HEP	$62K	$112K	$56K	$99K	$132K	>$500K	Expert Panel	Not Cost-Beneficial	Cost will exceed benefit.
186	Develop a procedure and obtain equipment to provide a temporary hookup of fire water to the RHR heat exchangers to use as a backup to CCW for removing decay heat.	Backup method of removing decay heat if CCW is lost.	SW02	$636K	$1M	$575K	$879K	$1.3M	>$1M	Expert Panel	Potentially Cost Beneficial	Ability to do this will require larger fire pumps

Table 9-1. Callaway Plant Potentially Cost Beneficial SAMAs

Callaway SAMA Number	Potential Improvement	Discussion	Additional Discussion
29	Provide capability for alternate injection via diesel-driven fire pump.	Improved injection capability.	Currently being evaluated by plant improvement program. Would use unborated water and portable pump (fire truck). Calculation of specific benefit of this SAMA was not performed since it is judged to be potentially low cost. Evaluation will consider impacts of injection of non-borated water.
64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie.	Improved ability to cool residual heat removal heat exchangers.	Cost based on development of procedure for temporary hookup of fire water to CCW heat exchangers. Cost of permanent modification would be much greater.
80	Provide a redundant train or means of ventilation.	Increased availability of components dependent on room cooling.	Procedures to open doors or provide temporary ventilation may be cost beneficial for the EDGs, MDAFW pumps, and charging pumps. Procedures for opening doors to the DC switchgear rooms exist.
160	Modifications to lessen impact of internal flooding path through Control Building dumbwaiter.	Lower impact of flood that propagates through the dumbwaiter	
162	Install a large volume EDG fuel oil tank at an elevation greater than the EDG fuel oil day tanks.	Allows transfer of EDG fuel oil to the EDG day tanks on failure of the fuel oil transfer pumps.	
178	Improvements to UHS cooling tower electrical room HVAC.	Improve availability or mitigate loss of HVAC.	Implementation of temporary ventilation or opening of doors will be a lower cost than the calculated benefit.

Table 9-1. Callaway Plant Potentially Cost Beneficial SAMAs (continued)

179	Modify procedures such that the water loop seals in the RCS cold legs are not cleared following core damage.	Prevents possible thermally induced steam generator tube rupture following core damage.	Implementation of procedure change will be lower cost than benefit, especially if 95% CDF benefit is considered.
180	Install lower amperage fuses for various 14 AWG control circuits in the MCR. The majority of the modification centers around the trip circuit fuses on NB, NG, PA, PB, and PG system breakers.	Reduced fire risk.	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
181	Install redundant fuses and isolation switches for MCR evacuation procedure OTO-ZZ-00001.	Reduced fire risk.	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
182	To protect against multiple spurious operation scenarios, cable runs will be changed to run a single wire in a protected metal jacket such that spurious valve opening due to a hot short affecting the valve control circuit is eliminated for the fire area. This modification will be implemented in multiple fire areas.	Reduced fire risk.	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
183	Quick response sprinkler heads in cable chases A-11, C-30, and C-31 will be modified to be in accordance with the applicable requirements of NFPA 13-1976 edition.	Reduced fire risk.	SAMA considered potentially cost beneficial without benefit or cost determination since the NFPA 805 license amendment request committed to performing the modification.
186	Develop a procedure and obtain equipment to provide a temporary hookup of fire water as a replacement for ESW	Backup cooling water if ESW/SW is lost	Ability to do this will require larger fire pumps

Table 11-1. Callaway Plant Release Category Frequency Results Obtained From SAMA Cases

RELEASE CATEGORY	BASE	NOATWS	INSTAIR	NOLOSP	NOSLOCA	CCW01	FW01	NOSGTR	NOSLB	CHG01
LERF-IS	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07
LERF-CI	1.658E-10	1.411E-10	1.658E-10	1.422E-10	6.210E-11	1.567E-10	1.658E-10	1.658E-10	1.610E-10	1.658E-10
LERF-CF	1.125E-08	1.103E-08	1.124E-08	7.372E-09	5.378E-09	1.071E-08	1.115E-08	1.135E-08	1.116E-08	1.123E-08
LERF-SG	2.331E-06	2.306E-06	2.330E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	0.000E+00	2.331E-06	2.331E-06
LERF-ITR	2.170E-07	1.845E-07	2.167E-07	1.309E-07	2.072E-07	2.170E-07	2.052E-07	0.000E+00	1.936E-07	2.169E-07
LATE-BMT	2.551E-06	2.268E-06	2.547E-06	1.254E-07	2.022E-06	2.507E-06	2.448E-06	2.626E-06	2.515E-06	2.467E-06
LATE-COP	3.185E-06	3.185E-06	3.185E-06	1.796E-08	3.170E-06	3.185E-06	3.185E-06	2.234E-06	3.185E-06	3.185E-06
SERF	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
INTACT	8.080E-06	8.075E-06	8.080E-06	7.065E-06	2.540E-06	7.573E-06	7.983E-06	8.119E-06	7.773E-06	8.137E-06
TOTAL	1.655E-05	1.620E-05	1.654E-05	9.851E-06	1.045E-05	1.600E-05	1.634E-05	1.316E-05	1.618E-05	1.652E-05

Table 11-1. Callaway Plant Release Category Frequency Results Obtained From SAMA Cases (Cont.)

RELEASE CATEGORY	SW01	NOSBO	LOCA05	H2BURN	RCPLOCA	LOCA 12	CONT02	LOCA04	LOCA03	CONT01
LERF-IS	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07
LERF-CI	1.658E-10	1.658E-10	6.210E-11	1.658E-10	1.567E-10	1.658E-10	0.000E+00	1.658E-10	1.658E-10	1.658E-10
LERF-CF	1.124E-08	1.030E-08	5.018E-09	4.102E-12	1.048E-08	1.099E-08	1.125E-08	1.114E-08	1.089E-08	1.125E-08
LERF-SG	2.331E-06	2.329E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.298E-06	2.331E-06
LERF-ITR	2.170E-07	1.443E-07	2.072E-07	2.170E-07	2.170E-07	2.165E-07	2.170E-07	2.170E-07	2.169E-07	2.170E-07
LATE-BMT	2.553E-06	1.611E-06	2.009E-06	2.551E-06	2.475E-06	1.893E-06	2.551E-06	2.441E-06	2.007E-06	2.551E-06
LATE-COP	3.181E-06	2.426E-06	3.170E-06	3.170E-06	3.173E-06	3.182E-06	3.185E-06	3.185E-06	3.185E-06	0.000E+00
SERF	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
INTACT	8.080E-06	7.883E-06	2.170E-06	8.080E-06	7.301E-06	8.329E-06	8.080E-06	8.080E-06	8.180E-06	8.080E-06
TOTAL	1.655E-05	1.458E-05	1.007E-05	1.652E-05	1.568E-05	1.614E-05	1.655E-05	1.644E-05	1.607E-05	1.336E-05

Table 11-1. Callaway Plant Release Category Frequency Results Obtained From SAMA Cases (Cont.)

RELEASE CATEGORY	BREAKER	DC01	SW02	CCW02	CST01	ISLOCA	LOSP1	DEPRESS	LOCA06	HVAC
LERF-IS	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	0.000E+00	1.730E-07	1.730E-07	1.730E-07	1.730E-07
LERF-CI	1.666E-10	1.658E-10	1.514E-10	1.422E-10	1.650E-10	1.658E-10	1.666E-10	1.658E-10	1.658E-10	1.658E-10
LERF-CF	1.129E-08	1.124E-08	9.088E-09	8.906E-09	1.112E-08	1.125E-08	1.113E-08	1.122E-08	1.109E-08	1.099E-08
LERF-SG	2.328E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.331E-06	2.329E-06
LERF-ITR	2.093E-07	2.170E-07	2.013E-07	2.108E-07	2.169E-07	2.170E-07	1.814E-07	2.160E-07	2.169E-07	1.944E-07
LATE-BMT	2.047E-06	2.551E-06	2.213E-06	1.864E-06	2.022E-06	2.551E-06	2.039E-06	2.508E-06	2.020E-06	1.657E-06
LATE-COP	3.210E-06	3.185E-06	8.964E-07	1.455E-06	3.185E-06	3.185E-06	2.991E-06	3.166E-06	3.185E-06	2.917E-06
SERF	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
INTACT	8.180E-06	8.080E-06	7.898E-06	7.836E-06	8.471E-06	8.080E-06	8.431E-06	8.069E-06	8.431E-06	8.312E-06
TOTAL	1.616E-05	1.655E-05	1.372E-05	1.388E-05	1.641E-05	1.638E-05	1.616E-05	1.647E-05	1.637E-05	1.559E-05

Table 11-1. Callaway Plant Release Category Frequency Results Obtained From SAMA Cases (Cont.)

RELEASE CATEGORY	FB01	PORV	EDGFUEL	FW02	SW03	HVAC02	RAI7a	SLIS	HEP	FWCCW2
LERF-IS	1.730E-07	1.730E-07	1.730E-10	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07	1.730E-07
LERF-CI	1.658E-10	1.658E-10	1.658E-10	1.658E-10	1.514E-10	1.658E-10	1.658E-10	1.658E-10	1.658E-10	1.567E-10
LERF-CF	1.094E-08	1.112E-08	1.124E-08	1.047E-08	1.031E-08	1.096E-08	1.135E-08	1.123E-08	1.080E-08	1.048E-10
LERF-SG	2.326E-06	2.331E-06	2.331E-06	2.324E-06	2.331E-06	2.331E-06	2.331E-06	2.290E-06	2.329E-06	2.317E-06
LERF-ITR	1.796E-07	2.169E-07	2.169E-07	1.659E-07	2.141E-07	2.169E-07	7.508E-08	2.138E-07	2.170E-07	2.170E-07
LATE-BMT	2.006E-06	2.022E-06	2.544E-06	1.983E-06	2.428E-06	1.990E-06	2.631E-06	2.545E-06	2.523E-06	2.467E-06
LATE-COP	3.185E-06	3.185E-06	3.182E-06	3.185E-06	2.557E-06	2.823E-06	3.235E-06	3.185E-06	3.185E-06	3.174E-06
SERF	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
INTACT	8.146E-06	8.471E-06	8.078E-06	7.796E-06	7.907E-06	8.461E-06	8.119E-06	8.036E-06	7.529E-06	7.311E-06
TOTAL	1.603E-05	1.641E-05	1.636E-05	1.564E-05	1.562E-05	1.601E-05	1.658E-05	1.645E-05	1.597E-05	1.566E-05

Excerpts from the Callaway Level 2 Analysis, Rev. 0

Section 2.2 Containment Event Tree Structure
Section 2.5 Release Categories
Section 3.1 Source Term Calculations

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2.2 CONTAINMENT EVENT TREE STRUCTURE

To assess the accident progression following a core damage event, this Level 2 analysis uses the containment event trees shown in Figures 2-1 & 2-2 based on the containment event trees provided in WCAP-16341-P. One event tree models station blackout scenarios, while the other models all other events. The event trees begin with one or a group of core damage sequences, then ask a number of questions to determine the type of release, if any, that occurs. Each question is modeled as a top event in the event tree and the outcome is based on previous work for Callaway, recent accident progression research, and the guidance provided in the WCAP. Each top event in the event trees is discussed below. Endstates on the event trees are discussed in Sections 2.4 and 2.5.

When implemented within the WinNUPRA model, these event trees are customized for each plant damage state, resulting in a large number of similar, but unique, containment event trees for quantification. Each unique CET is constructed by eliminating the unnecessary portions of the more general CETs, thereby maintaining the same structure and sequence labeling.

Plant Damage State Sequences

This first node of each containment event tree represents the collection of all core damage sequences from the Level 1 PRA into plant damage states. Station blackout sequences are assigned to the SBO tree, while all other core damage sequences follow the non-SBO tree. The assignment of core damage sequences to plant damage states is discussed in Section 2.3.

Containment Bypassed

Level 1 PRA sequences with an unisolated, initiating steam generator tube rupture (SGTR) or an unisolated, interfacing systems LOCA (ISLOCA) will bypass containment. The analysis of ISLOCA scenarios is documented separately [6,7]. The probability of an ISLOCA is incorporated directly into the Level 2 model through basic event BI with a probability of 1.73E-07 based on that analysis.

For SGTR core damage scenarios, the analysis assumes that the steam generator relief valve will stick open once it passes superheated water with high-temperature fission products, providing a direct path to the atmosphere. SGTRs with bypass would occur if failures allow the leakage through the tube rupture to continue indefinitely. From the Callaway Level 1 analysis, SGTR core damage sequences that include successful isolation of the ruptured steam generator and successful operator actions to depressurize the RCS, cooldown the RCS, and stop the safety injection will not lead to an unisolated bypass condition. SGTR sequences with such failures (including low probability event tree branches that do not question the actions) are treated as containment bypass scenarios. Subsequent CET questions can determine whether each scenario is a large and/or early release. The decision on this branch is determined by the accident sequence characteristics from the Level 1 PRA, which is indicated by the plant damage state designation.

Containment Isolated

For non-bypass scenarios, the possibility of containment isolation failure exists to provide a fission product release path through containment. The existing Callaway Level 1 analysis provides the associated containment isolation system (CIS) fault trees. The Level 2 model directly incorporates the CIS fault tree model into this top event. The containment isolation system includes all potential penetration locations with pipe sizes greater than 1". Further details of the containment isolation system analysis are located in the Containment Isolation System section 3.2.19 of the IPE [8]. This top event is modeled by gate ISO-FAIL in the general event tree, and by CI-XXX, where XXX represents the individual event trees as implemented in WinNUPRA.

Reactor Coolant System Pressure High

The next two top events have similar effects on accident progression, though the method by which it is achieved is different. The bottom branch of this top event, RCS Pressure High, represents core damage scenarios where the reactor coolant system is at low pressure due to a large (A) or medium (S1) loss of coolant accident or other scenarios with open relief valves that would depressurize the reactor. Low pressure means that pressure is insufficient to challenge the steam generator tubes or result in direct containment heating later in the accident progression (generally less than about 200 psi). The upper branch represents all other scenarios that will not depressurize through the break or relief valve. The decision on this branch is determined by the accident sequence characteristics from the Level 1 PRA, which is indicated by the plant damage state designation.

Steam Generator Feedwater Available

Another method for reducing reactor pressure is through use of the steam generators. If feedwater is available to the steam generators, decay heat is removed and the reactor can be reduced in pressure (to around 1000 psi). This pressure reduction will eliminate the challenge to the steam generator tubes and reduce the effects of direct containment heating (which is negligible for Callaway). The function of feedwater is modeled in the Level 1 PRA via both main feedwater and auxiliary feedwater and identified by the plant damage state.

Pressure-Induced Steam Generator Tube Rupture

Core damage sequences that continue on the high pressure branch are assumed to be at or near the primary PORV/SRV setpoint. Without water in the steam generators, there is a possibility of pressure-induced steam generator tube rupture early in the scenario. Because the pressure is high from the beginning of the scenario, this question is asked prior to any operator actions or other reactor coolant system failures that could depressurize the RCS. Details of this evaluation are based on WCAP-16341-P and shown in Appendix D. This event is modeled via basic event L2-SGT-VF-PISGR.

RCS Depressurize Early

If the steam generator tubes survive the initial pressure differential, the operators could take action to depressurize the reactor coolant system in order to reduce the likelihood of tube rupture or direct containment heating. To do so, the operators would open a primary system PORV. If successful, the scenario transfers to a low-pressure accident progression. If the RCS is not depressurized, either due to human inaction or equipment failure, additional high-pressure failures are considered. This action appears in the plant Severe Accident Control Room Guideline Initial Response SACRG-1 [9]. This top event is modeled by gate RCS-DEP1 in the general event tree, and by RD-XXX, where XXX represents the individual event trees as implemented in WinNUPRA. The gate couples the existing system fault tree G12P100, Failure of Both PORVs, from fault tree 12PORVS with an operator action OP-XHE-FO-DEP1, Operator Fails to Open PORV to Depressurize RCS. The human error probability for this operator action is set to 0.1 consistent with Section 4.8 of the IPE [10]. The effect of the operator action is examined in the sensitivity studies.

Thermally-Induced Steam Generator Tube Rupture

With the reactor coolant system remaining at high pressure and without feedwater to enough steam generators to depressurize the reactor, the likelihood of thermally-induced creep rupture of steam generator tubes is addressed. As with pressure-induced tube rupture, the age and condition of the steam generator tubes must be considered. Failure probabilities for moderately-damaged tubes are used to account for plant aging during the license renewal term. Details of this evaluation are shown in Appendix D along with the pressure-induced SGTR analysis. Basic event L2-SGT-VF-TISGR represents the probability in the model.

RCS Depressurize Late

During high-pressure core damage scenarios, a "race" occurs to determine where the RCS will first fail. While the reactor vessel will eventually fail as the molten core degrades the lower vessel head, failures may also occur in the steam generator tubes (discussed above) or in the hot leg or surge line of the reactor coolant system. For high-pressure, station-blackout-like scenarios which tend to occur on this branch, the likelihood of hot leg failure is very high. Based on the WCAP, this analysis uses a likelihood of 98% for hot leg failure (basic event L2-RCS-VF-DEP2). When hot leg failure occurs prior to vessel breach, the reactor coolant system depressurizes prior to failing the lower vessel head, thus eliminating the possibility of high-pressure core melt events leading to direct containment heating. This is generally a beneficial failure since it prevents direct containment heating.

Containment Failure Early

Three primary causes for containment failure at the time of reactor vessel breach apply to Callaway – steam explosion, hydrogen burn, and direct containment heating. The analysis of these containment challenges follows the guidance in WCAP-16341-P. Low pressure sequences (such as due to a LOCA) reduce reactor coolant system pressure to the point where containment is only subject to steam explosion and hydrogen burn challenges. Low pressure sequences due to steam generator cooling do not depressurize as far, but likewise only consider steam explosion and hydrogen burn since the already low threat due to direct containment heating would only decrease further. High pressure sequences with depressurization after core damage due to operator action or hot leg failure are primarily subject to hydrogen burn challenges. High pressure scenarios that remain at high pressure until the time of vessel breach are primarily subject to direct containment heating challenges, which includes the effects of hydrogen combustion and steam explosion. Therefore, different branches through the event tree require different early containment failure probabilities. This model assigns probability CFE1 to the combination of steam explosion and hydrogen burn, CFE3 to direct containment heating, and CFE5 to hydrogen burn alone. Recent research has provided an improved understanding of these phenomena and each is discussed below.

Ex-vessel steam explosions due to the pouring of the molten core into a pool of water can challenge the integrity of the containment via damage to the reactor cavity. Based on WCAP-16341-P, this is a greater issue for free-standing reactor cavities (as opposed to excavated cavities). Because Callaway is an excavated cavity, steam explosions do not pose a failure mechanism for early containment failure.

Hydrogen burns can challenge the integrity of the containment by creating high pressure excursions. The amount of hydrogen released into containment depends upon the amount of core damage at the time of vessel failure. Scenarios that lead to hydrogen burns at plants like Callaway are limited to about 50% zirconium oxidation for CFE5-type scenarios and 40% for CFE1-type scenarios. Based on WCAP-16341-P, the plant-specific probability of early containment failure at Callaway due to hydrogen burn is less than 0.001 at 40% oxidation and at 50% oxidation. To capture the possibility of containment failure due to hydrogen burn and/or steam explosion and maintain flexibility in the model, a probability of 0.001 will be used for both CFE1 and CFE5 in the model.

Direct containment heating is also addressed by WCAP-16341-P. The WCAP reports plant-specific conditional containment failure probabilities due to direct containment heating for several plants, including Callaway. The suggested probability is reported as 0.000 to cover all scenarios, and includes the effects of blowdown of the RCS, debris-to-gas heat transfer, exothermic metal/steam & metal/oxygen reactions, and hydrogen combustion that occur during a high-pressure melt ejection. To capture the possibility of DCH and maintain flexibility in the model, a CFE3 probability of 0.001 will be used in the model.

Containment Heat Removal

Containment Heat Removal can be accomplished through either the Containment Spray System (CS) or the Containment Coolers (VN). The Level 2 PRA models the containment heat removal function via gate NO-CHR in the general event tree based on the WCAP, and by CH-XXX, where XXX represents the individual event trees as implemented in WinNUPRA. The containment heat removal function consists of a combination of gates:

- GCS-100, Failure of CS Injection Mode, from existing system fault tree CS1
- GCSR100, Failure of CS Recirc Mode, from existing system fault tree CSREC
- GVN-100, Failure of VN, from existing system fault tree VN

Note that for some Level 2 scenarios, these functions may not be available due to power or cooling water failures, and the system fault trees model these support systems accordingly. Failure of containment heat removal will allow the containment to slowly pressurize until failure. The plant-specific MAAP calculations use a failure pressure of 134.9 psig to define containment overpressure failure [11].

No Large Early Release

For accidents that bypass containment or cause a containment isolation failure, it is possible that the release may be of insufficient magnitude to be classified as a large, early release.

For steam generator tube rupture scenarios that bypass containment and lead to an early release, the operators may still be able to reduce the magnitude of the release by providing feedwater to the ruptured steam generator in order to scrub fission products from the release. Such scrubbing should reduce the magnitude of the release such that it is no longer categorized as a large release. In the Level 2 model, this capability is included in the structure for future analysis, but is not currently credited. To credit such scrubbing as a fission product reduction mechanism, analysis would be required to include failures of secondary heat removal (i.e., feedwater to the steam generator) and human reliability analysis of the human action to keep the steam generator full. Success of the function would reduce the release to a small magnitude (non-LERF).

Note that some accidents initiated by a steam generator tube rupture may also have a relatively slow accident progression characterized by several hours between depletion of the refueling water storage tank and core uncovery. Depending on a plant's emergency response procedures, it is possible for the plant to make an anticipatory declaration of general emergency to allow more time for offsite protective actions. For SGTR sequences that do not have operable safety injection, this time delay will be much shorter, and an anticipatory declaration is not possible. At this time, no credit is taken for an anticipatory declaration in the Callaway model, but the action could be considered as an option to assess some early releases as late releases.

Containment isolation failures can also be assessed for their release magnitudes to determine whether they should be classified as LERF. Similarly to the SGTR scenarios, credit is not taken in the model for this distinction, though the Level 2 model structure does include the events to allow future evaluation if further analyses become available that support a distinction among containment isolation failures. Details of the containment isolation system analysis are located in the Containment Isolation System section 3.2.19 of the IPE [8].

Interfacing system LOCAs (ISLOCAs) could exist in locations that could fill with water and scrub the fission products from the release, thus providing another method of reducing a large release. However, based on the current ISLOCA analysis [6,7], such situations are not expected to occur for Callaway.

This branch of the containment event trees captures the possibility of a reduced or delayed release that would be classified as a non-LERF scenario. The Level 2 model provides for this possibility in the structure of the model, but does not credit such reductions in the current base model. The branches of the event tree are represented by basic events ISO-LG and BYP-LG.

Basemat Meltthrough

If no other containment failures occur during an accident scenario and containment heat removal exists, the last containment failure mode to examine is basemat meltthrough. If not cooled by an overlying water pool, the molten corium will begin to attack and erode the concrete basemat. Several beneficial factors at Callaway make basemat meltthrough less severe than other plants. First, Callaway has a "wet" containment design. If the RWST is injected into the primary system or containment via ECCS or containment spray, the water will drain to the reactor cavity and provide cooling of the molten corium, thus reducing the chance of basemat meltthrough. Second, the Callaway containment has a very thick basemat – 10 feet thick [12]. Even without cooling of the molten corium, basemat meltthrough will require many hours to erode through this thickness of concrete. Third, Callaway has a relatively large cavity floor area, meaning the molten corium will have more space to spread, resulting in a shallow layer (less than 1 foot) of corium which can be more easily cooled by overlying water. For the containment event trees, sequences including injection of the RWST can avoid basemat meltthrough with a high probability of success, while sequences without injection are subject to eventual basemat meltthrough. Because basemat meltthrough is only questioned if containment heat removal is successful, a wet cavity will be maintained if it was initially wet. The probability of having basemat meltthrough with a wet cavity is assigned a value of 0.05 (basic event L2-CNT-VF-BMMTW), based on guidance in the WCAP for a wet cavity with a shallow corium layer. For scenarios where the cavity is dry, basic event L2-CNT-VF-BMMTD models eventual basemat meltthrough with a probability of 1.0.

Core Damage Arrested Prior to Vessel Breach

WCAP-16341-P provides guidance to allow incorporation of actions to recover offsite power and restore injection during station blackout scenarios. As noted in the Assumptions, this analysis does not provide credit for recovery of offsite power after core damage but before a radioactive release. Given that power recovery has not occurred prior to core damage, there is a small, but non-zero chance of power recovery in the period between core damage and radioactive release. In addition, further recovery actions may be required to restore safety functions to arrest core damage and prevent a significant release. The time window available for these actions will vary for different scenarios, and therefore the slightly conservative assumption of no power recovery during this window is taken. The bottom failure branch is always followed for the Callaway Level 2 model, and is labeled in the general CET with the type of containment failure that could occur (either VB-low for low-pressure vessel breach or HPME for high-pressure melt ejection).

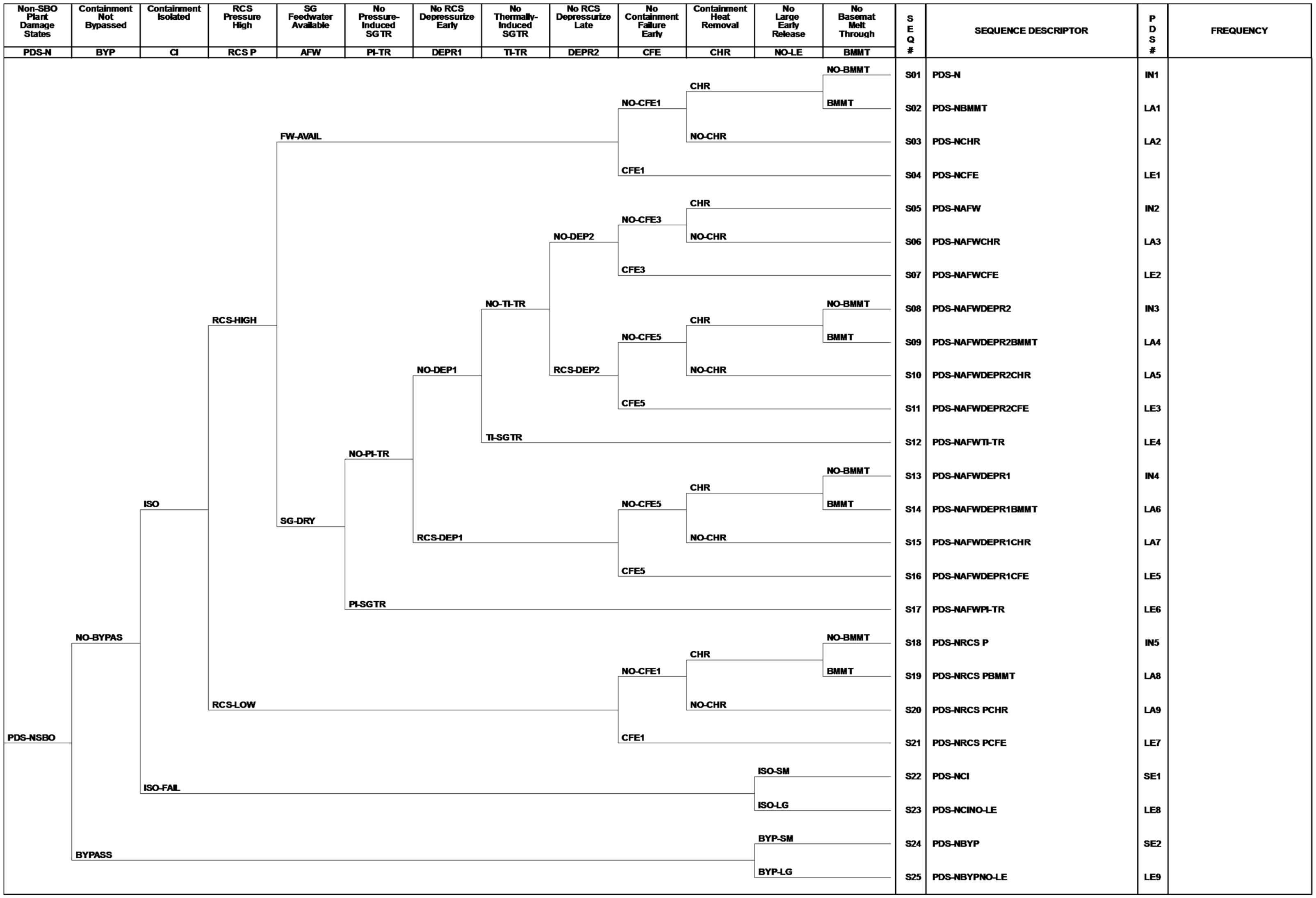

Figure 2-1

GENERAL CET FOR NON-SBO SCENARIOS

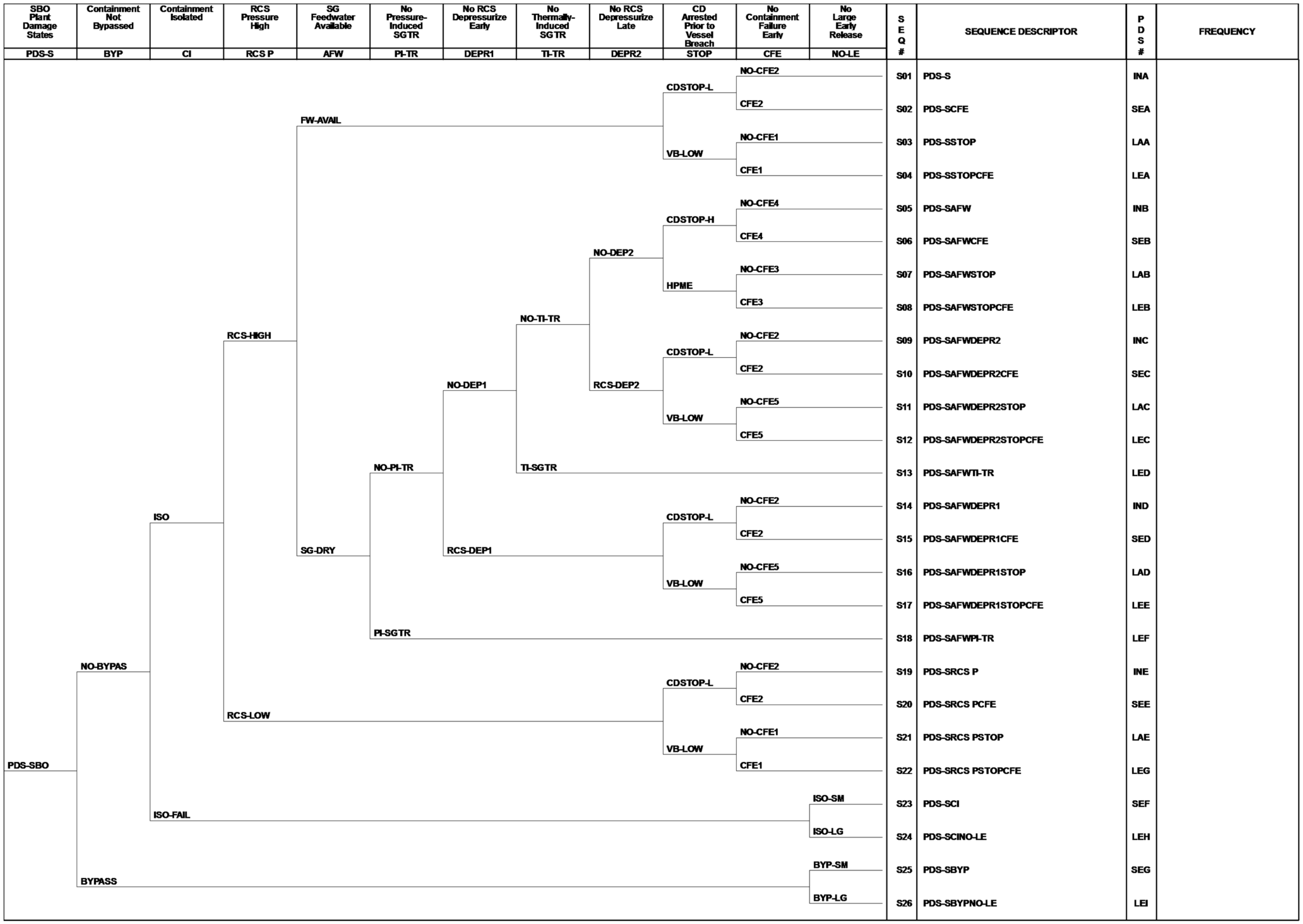

Figure 2-2

GENERAL CET FOR SBO SCENARIOS

2.5 RELEASE CATEGORIES

2.5.1 General Release Categories

As indicated in the previous section, the Level 2 PRA containment event tree sequences are categorized into four general release categories, which are described below.

INTACT

Containment structure and function succeed and prevent a substantial release of fission products. Source term calculations assume normal plant leakage to determine offsite consequences.

LATE

Containment failure occurs, but is considered late because of a significant time delay between core damage and containment failure. Releases may be large or small, but offsite consequences are limited to latent health effects and contamination.

SERF

Containment function is bypassed, but the radioactive release is scrubbed by an overlying water pool or limited by the size of the containment failure, reducing the offsite health effects.

LERF

Containment failure occurs early in the scenario. Early releases are defined as those releases that occur within a short time following core damage based on plant-specific source term calculations, such that adequate evacuation time is not available to protect the public from prompt health effects. Large releases are determined by plant-specific source term calculations.

2.5.2 Detailed Release Categories

A number of different Level 2 sequences contribute to each of the four general release categories above. Because the actual release characteristics will vary depending on how the containment event tree progresses, detailed release categories further define the Level 2 sequences. These detailed release categories consider the scenario characteristics and the ultimate containment failure mode. Each Level 2 sequence is mapped into one of these detailed release categories.

INTACT

This release category captures all of the INTACT sequences. Because the containment is essentially intact, sequence variations have a negligible impact on the release characteristics. INTACT-01, INTACT-02, INTACT-03, INTACT-04, and INTACT-05 contribute to this category. Releases to the environment are via normal containment leakage.

LATE-BMT

This release category captures sequences that result in basemat meltthrough. Because basemat meltthrough takes a significant amount of time to erode the thick basemat at Callaway, the release is small and significantly delayed. LATE-01, LATE-04, LATE-06, and LATE-08 contribute to this category.

LATE-COP

This release category captures sequences that result in containment failure due to late overpressure. LATE-02, LATE-03, LATE-05, LATE-07, LATE-09, LATE-B, LATE-C, and LATE-D contribute to this category.

LERF-IS

This release category captures sequences caused by an unisolated ISLOCA. Those sequences from LERF-09 with ISLOCA initiating events contribute to this category.

LERF-CI

This release category captures sequences that result in containment isolation failure due to either valve failure or excessive pre-existing containment leakage. LERF-08 and LERF-H contribute to this release category.

LERF-CF

This release category captures sequences that result in early containment failure due to steam explosion, hydrogen burn, and/or direct containment heating at the time of vessel breach. LERF-01, LERF-02, LERF-03, LERF-05, LERF-07, LERF-B, LERF-C, and LERF-E contribute to this category. Note that no credit for containment heat removal is credited as the function would not prevent containment failure, though it could affect ex-vessel cooling of the core if containment sprays fill up the reactor cavity.

LERF-SG

This release category captures sequences caused by a steam generator tube rupture. SGTR sequences with core damage provide a direct release path to the environment through the steam generator relief valves. Those sequences from LERF-09 with SGTR initiating events contribute to this category.

LERF-ITR

This release category captures sequences that result in either a pressure-induced or thermally-induced steam generator tube rupture that bypasses containment. LERF-04, LERF-06, LERF-D, and LERF-F contribute to this category.

3.1 SOURCE TERM CALCULATIONS

3.1.1 Representative Sequence Selection

For each detailed release category defined above, accident progression calculations predict the timing and amount of release. Because each release category can contain a high number of sequences, representative sequences must be defined for each category. For the INTACT, LATE, and SERF categories, the most likely contributing sequences are chosen to represent the category. For the LERF categories, both the likelihood of the scenario and its potential offsite effect is considered in order to capture the effects of all of the most likely scenarios within the category. The table below describes the representative sequences for each detailed release category. The first column includes the dominant Level 2 sequence to each release category, with the percentage of that category that the sequence contributes.

For INTACT sequences, containment structure and function succeed and prevent a large or late release of fission products. Source term calculations assume normal plant leakage to determine offsite consequences.

For LATE sequences, containment failure occurs, but is considered late because of a significant time delay between core damage and containment failure. Releases may be large or small, but offsite consequences are limited to latent health effects and contamination.

For SERF sequences, containment bypass occurs via a steam generator tube rupture, but the release is scrubbed to significantly reduce the offsite health effects.

For LERF sequences, containment failure occurs early in the scenario. Early releases are defined as those releases that occur within a short time following core damage, such that adequate evacuation time is not available to protect the public from prompt health effects. The magnitude of each release is determined by a plant-specific source term calculation.

Table 3-1
REPRESENTATIVE RELEASE SCENARIOS

Release Category	Dominant L2 Sequences	Dominant L1 Sequences	Representative Sequence
LERF-IS	LE9:100%	BI:100%	Large break ISLOCA through RHR cold leg, rupture in RHR HX, ECCS success, op fail to depressurize RCS & fail to refill RWST
LERF-CI	LE8:62% LEH:38%	T1S-S10:34% S2-S03:27% T1TC-S02:13%	LOOP/SBO, AFW avail for 8 hrs, 21gpm seal LOCAs, successful initial cooldown & depressurize, fail to restore power, cont isolation failure at t=0
LERF-CF	LEC:41% LE1:24% LE3:17% LE7:11%	T1S-S10:34% S2-S03:22% T1TC-S02:12%	LOOP/SBO, AFW avail for 8 hrs, 21gpm seal LOCAs, successful initial cooldown & depressurize, fail to restore power, cont failure at vessel breach
LERF-SG	LE9:100%	TSG-S07:85% TSG-S09:11%	SGTR, AFW to unbroken SGs, ECCS success, isolation of broken SG MSIV, op fail to cooldown/depressurize/stop SI, stuck-open SGRV when passing water
LERF-ITR	LED:49% LE4:22% LEF:20%	T1S-S10:61%	LOOP/SBO, AFW avail for 8 hrs, 21gpm seal LOCAs, successful initial cooldown & depressurize, fail to restore power, tube rupture at core damage
LATE-BMT	LA4:55% LA1:35%	T1S-S03: 20% T3-S06: 14% T2-S05: 11% TSW-S23: 11%	LOOP/SBO, AFW avail, 21gpm seal LOCAs, successful initial cooldown & depressurize, power restored at 8hrs, but failure of ECCS injection, CHR successful, cont failure by basemat meltthrough
LATE-COP	LAC:78% LAD:12%	T1S-S10:79%	LOOP/SBO, AFW avail for 8 hrs, 21gpm seal LOCAs, successful initial cooldown & depressurize, fail to restore power, CHR fail due to SBO, cont failure by overpressure @ 134.9 psig
SERF	NA*	NA*	Same as LERF-SG, but add AFW to ruptured SG at CD
INTACT	IN1:47% IN5:33% IN3:17%	S2-S03: 42% T1TC-S02: 23%	S2 LOCA, AFW successful, ECCS successful, fail to recirc, no cont faiure

*With current model, SERF does not occur, but release category is maintained for insights.
Percentage contributions based on Update 4 model.

Note that, in order to determine the dominant Level 1 sequences in the table above, the Level 1 eqn files were modified to add flags indicating which Level 1 sequence created each cutset. As these propagate through the Level 2 model with the attached flags, some non-minimal cutsets may occur in the Level 2 results, leading to a very slightly conservative result. These changes are not significant enough to affect the contribution percentages recorded in the table.

Response to RAIs 5e and 5f

Revised Tables:

Table 3-2: Level 1 Importance List Review

Table 3-6: LERF Importance Review

Table 3-7: Late Release Importance Review

Table 3-2. Level 1 Importance List Review

Basic Event Name	**Basic Event Description**	**RRW**	**Associated SAMA**
IE-S2	SMALL LOCA INITIATING EVENT FREQUENCY	1.554	25-42
IE-T1	LOSS OF OFFSITE POWER INITIATING EVENT FREQUENCY	1.514	1-24
OP-XHE-FO-ECLRS2	OPERATOR FAILS TO ALIGN ECCS SYSTEMS FOR COLD LEG RECIRC	1.389	36
IE-TSG	STEAM GENERATOR TUBE RUPTURE IE FREQUENCY	1.166	119-129
OP-XHE-FO-SGTRDP	OPERATOR FAILS TO C/D AND DEPRESS THERCS AFTER SGTR	1.082	see note on operator action events
OP-XHE-FO-SGTRWR	OPERATOR FAILS TO C/D AND DEPRESS RCSAFTER WATER RELIEF	1.082	see note on operator action events
IE-T3	TURBINE TRIP WITH MAIN FEEDWATER AVAILABLE IE FREQ	1.07	26, 28-33, 36, 39, 41-46, 49-64, 70, 72, 79, 80, 82, 83, 89-103, 107, 110, 149, 161, 163, 166-169, 171, 172, 175-176, 178, 179 provide mitigation of possible impacts.

Table 3-2. Level 1 Importance List Review (continued)

BB-PRV-CC-V455A	PRESSURIZER PORV PCV455A FAILS TO OPEN	1.053	89
BB-PRV-CC-V456A	PRESSURIZER PORV PCV456A FAILS TO OPEN	1.053	89
NE-DGN-DR-NE01-2	DGNS CC FTR.	1.049	1-24
AE-CKV-DF-V120-3	CHECK VALVES AEV120121,122,123 COMMON CAUSE FAIL TO OPEN	1.048	163
EF-PSF-TM-ESWTNB	ESW TRAIN B IN TEST OR MAINTENANCE	1.045	46-57, 62-64
OP-XHE-FO-ACRECV	OPERATOR FAILS TO RECOVER FROM A LOSSOF OFFSITE POWER	1.044	22
EF-PSF-TM-ESWTNA	ESW TRAIN A IN TEST OR MAINTENANCE	1.043	46-57, 62-64
FAILTORECOVER-8	PROBABILITY THAT POWER IS NOT RECOV-ERED IN 8 HOURS.	1.042	1-24
EF-MDP-DR-EFPMPS	ESW PUMPS CC FTR.	1.041	46-57, 62-64
OP-XHE-FO-CCWRHX	OPERATOR FAILS TO INITIATE CCW FLOW TO THE RHR HXS	1.037	185
FAILTORECOVER-12	CONDITIONAL PROB. THAT PWR IS NOT RE-COVERED IN 12 HRS.	1.035	1-24
EF-MDP-FR-PEF01A	ESW PUMP A (PEF01A)FAILS TO RUN	1.033	46-57, 62-64
FB-XHE-FO-FANDB	OPERATOR FAILS TO ESTABLISH RCS FEED AND BLEED	1.032	see note on operator action events
OP-XHE-FO-ECLR	OPERATOR FAILS TO ALIGN ECCS SYSTEMS FOR COLD LEG RECIRC	1.031	36
TORNADO-T1-EVENT	CONDITIONAL PROB. TORNADO T(1) EVENT LOSS OF AEPS	1.031	15
EF-MDP-FR-PEF01B	ESW PUMP B (PEF01B)FAILS TO RUN	1.025	46-57, 62-64
EG-MDP-DS-EGPMP4	ALL 4 EG PUMPS CC FTS.	1.025	59
IE-S1	INTERMEDIATE LOCA INITIATING EVENT FREQUENCY	1.023	25-42
IE-TMSO	MAIN STEAMLINE BREAK OUTSIDE CTMT IE FREQUENCY	1.022	153

Table 3-2. Level 1 Importance List Review (continued)

AL-TDP-TM-TDAFP	TDAFP IN TEST OR MAINTENANCE	1.019	66, 68, 75, 78
BB-RCA-WW-RCCAS	TWO OR MORE RCCA'S FAIL TO INSERT (MECH. CAUSES)	1.019	130-137
EF-DRAIN-TRAINB	ALL TRAIN B SW UNAVAIL. DUE TO DRAINAGE OF EF TRAIN B.	1.019	46-57, 62-64
EG-HTX-TM-CCWHXB	CCW TRAIN B TEST/MAINT. (E.G. HX B TEST/MAINT.)	1.016	59
VL-ACX-DS-GL10AB	ROOM COOLER SGL10A, B CC FTS	1.014	80
EF-MOV-CC-EFHV37	VALVE EFHV37 FAILS TO OPEN	1.013	46-57, 62-64
IE-S3	VERY SMALL LOCA INITIATING EVNET	1.013	25-42
NE-DGN-FR-NE0112	DIESEL GENERATOR NE01 FTR - 12 HR MT	1.013	1-24
NE-DGN-FR-NE0212	DIESEL GENERATOR NE02 FTR - 12 HR MT	1.013	1-24
NE-DGN-TM-NE01	DIESEL GENERATOR NE01 IN TEST OR MAINTENANCE	1.013	1-24
NE-DGN-TM-NE02	DIESEL GENERATOR NE02 IN TEST OR MAINTENANCE	1.013	1-24
IE-T2	LOSS OF MAIN FEEDWATER IE FREQUENCY	1.012	65-79
NE-DGN-FS-NE01	DIESEL GENERATOR NE01 FAILS TO START	1.012	1-24
AL-TDP-FS-TDAFP	TDAFP FAILS TO START	1.011	66, 68, 75, 78
EF-MDP-FS-PEF01A	ESW PUMP A (PEF01A)FAILS TO START	1.011	46-57, 62-64
EJ-PSF-TM-EJTRNB	RHR TRAIN B IN TEST OR MAINTENANCE	1.011	25-42
NE-DGN-FS-NE02	DIESEL GENERATOR NE02 FAILS TO START	1.011	1-24
EF-MDP-DS-EFPMPS	ESW PUMPS CC FTS	1.01	46-57, 62-64
EF-MOV-CC-EFHV38	VALVE EFHV38 FAILS TO OPEN	1.01	46-57, 62-64
OP-XHE-FO-AEPS1	OPERATOR FAILS TO ALIGN AEPS TO NB BUS IN 1 HR	1.01	1-24
VD-FAN-FR-CGD02A	UHS C.T. ELEC. ROOM SUPPLY FAN CGD02A FAILS TO RUN	1.01	178

Table 3-2. Level 1 Importance List Review (continued)

AE-CKV-DF-V124-7	CHECK VALVES AEV124,125,126,127 COMMON CAUSE FAIL TO OPEN	1.009	163
AEPS-ALIGN-NB02	PDG ALIGN TO NB02 (FAIL TO ALIGN PDG TO NB02)	1.009	1-24
EF-MDP-FS-PEF01B	ESW PUMP B (PEF01B)FAILS TO START	1.009	46-57, 62-64
EF-MOV-D2-V37-38	VALVES EFHV37 & 38 COMMON CAUSE FAIL TO CLOSE (2 VALVES)	1.009	46-57, 62-64
FAILTOMNLINSRODS	OPERATOR FAILS TO MANUALLY DRIVE RODS INTO CORE	1.009	see note on operator action events
OP-COG-FRH1	OPERATORS FAIL TO DIAGNOSE RED PATH ON HEAT SINK	1.009	see note on operator action events
VD-FAN-FR-CGD02B	UHS C.T. ELEC. ROOM SUPPLY FAN CGD02B FAILS TO RUN	1.009	178
AEPS-ALIGN-NB01	PDG ALIGN TO NB01 (FAIL TO ALIGN PDG TO NB01)	1.008	1-24
AL-XHE-FO-SBOSGL	OPERATOR FAILS TO CONTROL S//G LEVEN AFTER COMPLEX EVENT	1.008	see note on operator action events
EF-MOV-OO-EFHV59	VALVE EFHV59 FAILS TO CLOSE	1.008	46-57, 62-64
EJ-PSF-TM-EJTRNA	RHR TRAIN A IN TEST OR MAINTENANCE	1.008	25-42
FAILTOREC-EFHV59	OPERATORS FAIL TO RECOVER (CLOSE) EFHV59	1.008	see note on operator action events
VL-ACX-FS-SGL10A	ROOM COOLER FAN SGL10A FAILS TO START	1.008	80
AL-PSF-TM-ALTRNB	AFW TRAIN B IN TEST OR MAINTENANCE	1.007	66, 68, 75, 78
BN-TNK-FC-RWSTUA	RWST UNAVAILALBE	1.007	171
EG-MDP-DR-EGPMP4	ALL 4 EG PUMPS CC FTR.	1.007	54-59, 61, 63, 64
EJ-XHE-FO-PEJ01	OPERATOR FAILS TO START AN RHR PUMP FOR LONG TERM C/D	1.007	see note on operator action events

Table 3-2. Level 1 Importance List Review (continued)

IE-TC	LOSS OF ALL COMPONENT COOLING WATER IE FREQUENCY	1.007	54-59, 61, 63, 64
IE-TSW	LOSS OF SERVICE WATER INITIATING EVENT	1.007	46-57, 62-64
SA-ICC-AF-RWSTL1	NO INPUT FOR RX TRIP FROM RPS	1.007	130-137
AE-XHE-FO-MFWFLO	FAILURE TO RE-ESTABLISH MFW FLOW DUE TO HUMAN ERRORS	1.006	see note on operator action events
BG-MDP-FR-NCP	MOTOR DRIVEN CHARGING PUMP FAILS TO RUN	1.006	25-42
EJ-MDP-DS-EJPMPS	RHR PUMPS CC FAIL TO START	1.006	25-42
EJ-MOV-CC-V8811A	VALVE EJHV8811A FAILS TO OPEN	1.006	25-42
IE-TFLB	FEEDLINE BREAK DOWNSTREAM OF CKVS IE FREQUENCY	1.006	65-79
NF-ICC-AF-LSELSA	LOAD SHEDDER TRAIN A FAILS TO SHED LOADS	1.006	1-24
OP-XHE-FO-SGISO	OPERATOR FAILS TO ISOLATE THE FAULTED S/G FOLLOWING SGTR	1.006	see note on operator action events
SA-ICC-AF-MSLIS	NO SLIS ACTUATION SIGNAL	1.006	184
SA-ICC-AF-RWSTL4	NO RWST LOW LEVEL SIGNAL AVAILABLE (SEP GRP 4)	1.006	25-42
VL-ACX-FS-SGL10B	ROOM COOLER FAN SGL10B FAILS TO START	1.006	80
VM-BDD-CC-GMD001	DAMPER GMD001 FAILS TO OPEN	1.006	80
VM-BDD-CC-GMD004	DAMPER GMD004 FAILS TO OPEN	1.006	80
VM-EHD-CC-GMTZ1A	ELEC/HYDR OP DAMPER GMTZ01A FAILS TO OPEN	1.006	80
AL-MDP-FR-MDAFPB	MDAFPB FAILS TO RUN AFTER START	1.005	66, 68, 75, 78
AL-TDP-FR-TDAFP	TDAFP FAILS TO RUN AFTER START	1.005	66, 68, 75, 78
BM-AOV-OO-BMHV1	BLOWDOWN ISOLATION VALVE BMHV0001 FAILS TO CLOSE	1.005	66, 68, 75, 78
BM-AOV-OO-BMHV4	BLOWDOWN ISOLATION VALVE BMHV0004 FAILS TO CLOSE	1.005	66, 68, 75, 78

Table 3-2. Level 1 Importance List Review (continued)

EJ-MOV-CC-V8811B	VALVE EJHV8811B FAILS TO OPEN	1.005	25-42
EJ-MOV-D2-8811AB	VALVES EJHV8811A & B COMMON CAUSE FAIL TO OPEN	1.005	25-42
NE-DGN-FR-NE01-2	DGN NE01 FAILS TO RUN (1 HR MISSION TIME)	1.005	1-24
NF-ICC-AF-LSELSB	LOAD SHEDDER TRAIN B FAILS TO SHED LOADS	1.005	1-24
VM-BDD-CC-GMD006	DAMPER GMD006 FAILS TO OPEN	1.005	80
VM-BDD-CC-GMD009	DAMPER GMD009 FAILS TO OPEN	1.005	80
VM-EHD-CC-GMTZ11	ELEC/HYDR OP DAMPER GMTZ11A FAILS TO OPEN	1.005	80

RCS = reactor coolant system; IE = initiating event; CC = common cause; FTR = fail to run; ESW = essential service water; ECCS = emergency core cooling system; FTS = fail to start

Note 1 – The current plant procedures and training meet current industry standards. There are no additional specific procedure improvements that could be identified that would affect the result of the HEP calculations. Therefore, no SAMA items were added to the plant specific list of SAMAs as a result of the human actions on the list of basic events with RRW greater than 1.005.

Table 3-6. LERF Importance Review

Basic Event Name	Basic Event Description	RRW	Associated SAMA
IE-TSG	STEAM GENERATOR TUBE RUPTURE IE FREQUENCY	6.808	119-129
OP-XHE-FO-SGTRDP	OPERATOR FAILS TO C/D AND DEPRESS THERCS AFTER SGTR	1.835	See note on operator action events
OP-XHE-FO-SGTRWR	OPERATOR FAILS TO C/D AND DEPRESS RCSAFTER WATER RELIEF	1.835	See note on operator action events
BB-PRV-CC-V455A	PRESSURIZER PORV PCV455A FAILS TO OPEN	1.314	161
BB-PRV-CC-V456A	PRESSURIZER PORV PCV456A FAILS TO OPEN	1.314	161
BI	ISLOCA CDF	1.068	111-113
OP-XHE-FO-SGISO	OPERATOR FAILS TO ISOLATE THE FAULTEDS/G FOLLOWING SGTR	1.037	See note on operator action events
IE-T1	LOSS OF OFFSITE POWER INITIATING EVENT FREQUENCY	1.034	1-24
IE-T3	TURBINE TRIP WITH MAIN FEEDWATER AVAILABLE IE FREQ	1.028	26, 28-33, 36, 39, 41-46, 49-64, 70, 72, 79, 80, 82, 83, 89-103, 107, 110, 149, 161, 163, 166-169, 171, 172, 175-176, 178, 179 provide mitigation of possible impacts.
AB-ARV-DF-SGPRVS	S/G PORVS ABPV01, 02, 03, & 04 COMMONCAUSE FAIL TO OPEN	1.024	89
AB-ARV-TM-ABPV03	S/G PORV ABPV0003 ISOLATED FOR TEST/MAINTENANCE	1.024	89
FB-XHE-FO-FANDB	OPERATOR FAILS TO ESTABLISH RCS FEED AND BLEED	1.023	SAMA 36, see note on operator action events

Table 3-6. LERF Importance Review (continued)

AE-CKV-DF-V120-3	CHECK VALVES AEV120121,122,123 COMMON CAUSE FAIL TO OPEN	1.022	163
AB-ARV-TM-ABPV01	S/G PORV ABPV0001 ISOLATED FOR TEST/MAINTENANCE	1.02	89
BB-RCA-WW-RCCAS	TWO (2) OR MORE RCCA's FAIL TO IN- SERT (MECH. CAUSES)	1.02	130-137
SA-ICC-AF-MSLIS	NO SLIS ACTUATION SIGNAL	1.016	184
AB-ARV-TM-ABPV04	S/G PORV ABPV0004 ISOLATED FOR TEST/MAINTENANCE	1.015	89
AB-PHV-OO-ABHV17	MSIV "B" (AB-HV-17) FAILS TO CLOSE ON DEMAND	1.015	89
TORNADO-T1-EVENT	CONDITIONAL PROB. TORNADO T(1) EVENT LOSS OF AEPS	1.014	15
BB-RLY-FT-72455	72 RELAY FAILS TO TRANSFER	1.011	79
BB-RLY-FT-72456	72 RELAY FAILS TO TRANSFER	1.011	79
BB-RLY-FT-AR455	AUX. RELAY FAILS TO TRANSFER	1.011	79
BB-RLY-FT-AR456	AUX. RELAY FAILS TO TRANSFER	1.011	79
NE-DGN-DR-NE01-2	DGNS CC FTR.	1.01	1-24
AB-ARV-CC-ABPV04	S/G PORV ASPV0004 FAILS TO OPEN	1.009	89
VL-ACX-DS-GL10AB	ROOM COOLER SGL10A, B CC FTS	1.009	80
AB-ARV-CC-ABPV01	S/G PORV ASPV0001 FAILS TO OPEN	1.008	89
AE-XHE-FO-MFWFLO	FAILURE TO RE-ESTABLISH MFW FLOW DUE TO HUMAN ERRORS	1.008	See note on operator action events
AL-TDP-TM-TDAFP	TDAFP IN TEST OR MAINTENANCE	1.008	66, 68, 75, 78
IE-TMSO	MAIN STEAMLINE BREAK OUTSIDE CTMT IE FREQUENCY	1.008	153
AB-ARV-CC-ABPV03	S/G PORV ASPV0003 FAILS TO OPEN	1.007	89
NE-DGN-FR-NE0112	DIESEL GENERATOR NE01 FTR - 12 HR MT	1.007	1-24

Table 3-6. LERF Importance Review (continued)

NE-DGN-FR-NE0212	DIESEL GENERATOR NE02 FTR - 12 HR MT	1.007	1-24
EJ-PSF-TM-EJTRNB	RHR TRAIN B IN TEST OR MAINTENANCE	1.006	25-42
OP-XHE-FO-ECA32	OPERATOR FAILS TO PERFORM C/D TO COLD S/D IAW ECA 3.2	1.006	See note on operator action events
AB-AOV-CC-ABUV34	STEAM DUMP ABUV0034 FAILS TO OPEN	1.005	89
AB-AOV-CC-ABUV35	STEAM DUMP ABUV0035 FAILS TO OPEN	1.005	89
AB-AOV-CC-ABUV36	STEAM DUMP ABUV0036 FAILS TO OPEN	1.005	89
AL-XHE-FO-SBOSGL	OPERATOR FAILS TO CONTROL S//G LEVEN AFTER COMPLEX EVENT	1.005	See note on operator action events
EJ-XHE-FO-PEJ01	OPERATOR FAILS TO START AN RHR PUMP FOR LONG TERM C/D	1.005	See note on operator action events
FAILTOMNLINSRODS	OPERATOR FAILS TO MANUALLY DRIVE RODS INTO CORE	1.005	130-137

ISLOCA = interfacing system LOCA; S/G = steam generator

Note 1 – The current plant procedures and training meet current industry standards. There are no additional specific procedure improvements that could be identified that would affect the result of the HEP calculations. Therefore, no SAMA items were added to the plant specific list of SAMAs as a result of the human actions on the list of basic events with RRW greater than 1.005.

Table 3-7. Late Release Importance Review

Basic Event Name	Basic Event Description	RRW	Associated SAMA
IE-T1	LOSS OF OFFSITE POWER INITIATING EVENT FREQUENCY	4.51	1-24
RECSWT1	RECOVERY POWER AND SW IN 8 HRS BEFORE CORE UNCVRED	1.474	1-24
OP-XHE-FO-ACRECV	OPERATOR FAILS TO RECOVER FROM A LOSSOF OFFSITE POWER	1.14	SAMA 22, see note on operator action events
EF-PSF-TM-ESWTNB	ESW TRAIN B IN TEST OR MAINTENANCE	1.136	46-57, 62-64
NE-DGN-DR-NE01-2	DGNS CC FTR.	1.133	1-24
EF-MDP-DR-EFPMPS	ESW PUMPS CC FTR.	1.129	46-57, 62-64
EF-PSF-TM-ESWTNA	ESW TRAIN A IN TEST OR MAINTENANCE	1.127	46-57, 62-64
FAILTORECOVER-8	PROBABILITY THAT POWER IS NOT RECOV-ERED IN 8 HOURS.	1.105	1-24
FAILTORECOVER-12	CONDITIONAL PROB. THAT PWR IS NOT RE-COVERED IN 12 HRS.	1.098	1-24
IE-T3	TURBINE TRIP WITH MAIN FEEDWATER AVAILABLE IE FREQ	1.088	26, 28-33, 36, 39, 41-46, 49-64, 70, 72, 79, 80, 82, 83, 89-103, 107, 110, 149, 161, 163, 166-169, 171, 172, 175-176, 178, 179 provide mitigation of possible impacts.
EF-MDP-FR-PEF01A	ESW PUMP A (PEF01A)FAILS TO RUN	1.085	46-57, 62-64

Table 3-7. Late Release Importance Review (continued)

FB-XHE-FO-FANDB	OPERATOR FAILS TO ESTABLISH RCS FEED AND BLEED	1.076	SAMA 36, see note on operator action events
EF-MDP-FR-PEF01B	ESW PUMP B (PEF01B)FAILS TO RUN	1.074	46-57, 62-64
TORNADO-T1-EVENT	CONDITIONAL PROB. TORNADO T(1) EVENT LOSS OF TEMP EDGS	1.073	1-24
IE-S2	SMALL LOCA INITIATING EVENT FREQUENCY	1.067	25-42
AE-CKV-DF-V120-3	CHECK VALVES AEV120121,122,123 COMMON CAUSE FAIL TO OPEN	1.05	163
BB-RCA-WW-RCCAS	TWO (2) OR MORE RCCA's FAIL TO IN- SERT (MECH. CAUSES)	1.048	130-137
OP-XHE-FO-ECLRS2	OPERATOR FAILS TO ALIGN ECCS SYSTEMS FOR COLD LEG RECIRC	1.042	SAMA 36, see note on operator action events
EF-DRAIN-TRAINB	ALL TRAIN B SW UN- AVAIL. DUE TO DRAINAGE OF EF TRAIN B.	1.036	46-57, 62-64
NE-DGN-TM-NE02	DIESEL GEN NE02 IN TEST OR MAINTENANCE	1.034	1-24
NE-DGN-FR-NE0112	DIESEL GENERATOR NE01 FTR - 12HR MT	1.033	1-24
EF-MOV-CC-EFHV37	VALVE EFHV37 FAILS TO OPEN	1.032	46-57, 62-64
IE-S3	VERY SMALL LOCA INITIATING EVENT FREQUENCY	1.032	25-42
NE-DGN-FR-NE0212	DIESEL GENERATOR NE02 FTR - 12HR MT	1.032	1-24
NE-DGN-TM-NE01	DIESEL GEN NE01 IN TEST OR MAINTENANCE	1.032	1-24
NE-DGN-FS-NE01	DIESEL GENERATOR NE01 FAILS TO START	1.03	1-24
NON-TORNADO-T1	CONDITIONAL PROB. T(1) EVENT NOT CAUSED BY TORNADO	1.03	1-24
VD-FAN-FR-CGD02A	UHS C.T. ELEC. ROOMSUPPLY FAN CGD02A FAILS TO RUN	1.03	178

Table 3-7. Late Release Importance Review (continued)

NE-DGN-FS-NE02	DIESEL GENERATOR NE02 FAILS TO START	1.029	1-24
OP-XHE-FO-DEP1	Operator Fails to Open PORV to Depressurize RCS	1.029	See note on operator action events
EF-MDP-DS-EFPMPS	ESW PUMPS CC FTS.	1.028	46-57, 62-64
EF-MOV-CC-EFHV38	VALVE EFHV38 FAILS TO OPEN	1.028	46-57, 62-64
EF-MDP-FS-PEF01A	ESW PUMP A (PEF01A)FAILS TO START	1.027	46-57, 62-64
EF-MDP-FS-PEF01B	ESW PUMP B (PEF01B)FAILS TO START	1.027	46-57, 62-64
EF-MOV-D2-V37-38	COMMON CAUSE FAIL.-VALVES EF-HV-37 AND38 FTC.	1.027	46-57, 62-64
VD-FAN-FR-CGD02B	UHS C.T. ELEC. ROOMSUPPLY FAN CGD02B FAILS TO RUN	1.026	178
OP-XHE-FO-AEPS1	OPERATOR FAIL TO ALIGN AEPS TO NB BUS IN 1 HR	1.025	See note on operator action events
FAILTOMNLINSRODS	OPERATOR FAILS TO MANUALLY DRIVE RODSINTO CORE (RI).	1.023	130-137
EF-MOV-OO-EFHV59	VALVE EFHV59 FAILS TO CLOSE	1.022	46-57, 62-64
FAILTOREC-EFHV59	OPERATORS FAIL TO RECOVER (CLOSE) EFHV59.	1.022	See note on operator action events
BN-TNK-FC-RWSTUA	RWST UNAVAILABLE	1.02	171
AEPS-ALIGN-NB01	PDG ALIGN TO NB01 (FAIL TO ALIGN PDG TO NB01)	1.016	1-24
AEPS-ALIGN-NB02	PDG ALIGN TO NB02 (FAIL TO ALIGN PDG TO NB02)	1.015	1-24
AL-TDP-TM-TDAFP	TDAFP IN TEST OR MAINTENANCE	1.015	66, 68, 75, 78
IE-T2	LOSS OF MAIN FEEDWATER IE FREQUENCY	1.013	65-79
NF-ICC-AF-LSELSA	LOAD SHEDDER TRAIN A FAILS TO SHED LOADS	1.013	1-24

Table 3-7. Late Release Importance Review (continued)

NF-ICC-AF-LSELSB	LOAD SHEDDER TRAIN B FAILS TO SHED LOADS	1.013	1-29
VM-BDD-CC-GMD001	DAMPER GMD001 FAILS TO OPEN	1.013	80
VM-BDD-CC-GMD004	DAMPER GMD004 FAILS TO OPEN	1.013	80
VM-BDD-CC-GMD006	DAMPER GMD006 FAILS TO OPEN	1.013	80
VM-BDD-CC-GMD009	DAMPER GMD009 FAILS TO OPEN	1.013	80
VM-EHD-CC-GMTZ11	ELEC/HYDR OP DAMPER GMTZ11A FAILS TO OPEN	1.013	80
VM-EHD-CC-GMTZ1A	ELEC/HYDR OP DAMPER GMTZ01A FAILS TO OPEN	1.013	80
NE-DGN-FR-NE01-2	DGN NE02 FAILS TO RUN (1 HR MISSION TIME)	1.012	1-24
NE-DGN-FR-NE02-2	DGN NE02 FAILS TO RUN (1 HR MISSION TIME)	1.011	1-24
EF-CKV-DF-V01-04	CHECK VALVES EFV001 AND EFV004 COMMON CAUSE FAIL TO OPEN	1.009	46-57, 62-64
MANLRODINSERTION	OPERATORS MANUALLY DRIVE RODS INTO THE CORE	1.009	130-137
VM-FAN-FS-CGM01A	DIESEL GEN SUPPLY FAN CGM01A FAILS TO START	1.009	80
VM-FAN-FS-CGM01B	DIESEL GEN SUPPLY FAN CGM01B FAILS TO START	1.009	80
AE-CKV-DF-V124-7	CHECK VALVES AEV124,125,126,127 COMMON CAUSE FAIL TO OPEN	1.008	163
AE-XHE-FO-MFWFLO	FAILURE TO RE-ESTABLISH MFW FLOW DUE TO HUMAN ERRORS	1.008	See note on operator action events
EG-AOV-DF-TV2930	COMMON CAUSE FAILURE EG-TV-29 AND 30 TO CLOSE	1.008	46-57, 62-64
EG-HTX-TM-CCWHXB	CCW TRAIN B TEST/MAINT. (E.G. HX B TEST/MAINT.)	1.008	46-57, 62-64
IE-TFLB	FEEDLINE BREAK DOWNSTREAM OF CKVS IE FREQUENCY	1.008	65-79
AL-TDP-FS-TDAFP	TDAFP FAILS TO START	1.007	66, 68, 75, 78
AL-XHE-FO-SBOSGL	OPERATOR FAILS TO CONTROL S//G LEVEN AFTER COMPLEX EVENT	1.007	See note on operator action events

Table 3-7. Late Release Importance Review (continued)

IE-TSW	LOSS OF SERVICE WATER INITIATING EVENT	1.007	46-57, 62-64
NB-BKR-CC-NB0112	BREAKER NB0112 FAILS TO OPEN	1.007	1-24
NE-DGN-DS-NE01-2	DGNS CC FTS.	1.007	1-24
BG-MDP-TM-CCPA	CCP A IN TEST OR MAINTENANCE	1.006	25-42
BG-MDP-TM-CCPB	CCP B IN TEST OR MAINTENANCE	1.006	25-42
EG-MDP-DS-EGPMP4	ALL 4 EG PUMPS CC FTS.	1.006	25-42
IE-TMSO	MAIN STEAMLINE BREAK OUTSIDE CTMT IE FREQUENCY	1.006	153
NB-BKR-CC-NB0209	BREAKER NB0209 FAILS TO OPEN	1.006	1-24
VD-FAN-FS-CGD02A	UHS C.T. ELEC. ROOMSUPPLY FAN CGD02A FAILS TO START	1.006	178
IE-TDCNK01	LOSS OF VITAL DC BUS NK01 INITIATING EVENT FREQUENCY	1.005	3, 5, 6, 7,
OP-XHE-FO-CCWRHX	OPERATOR FAILS TO INITIATE CCW FLOW TO THE RHR HXS	1.005	185
OP-XHE-FO-ESW2HR	OPERATOR FAILS TO START AND ALIGN ESW 2 HR AFTER SW LOSS	1.005	See note on operator action events
VD-FAN-DR-GD02AB	FANS CGD02A,B COMMON CAUSE FTS	1.005	178
VD-FAN-FS-CGD02B	UHS C.T. ELEC. ROOMSUPPLY FAN CGD02B FAILS TO START	1.005	178
VM-FAN-DS-GMFANS	FANS CGM01A,B COMMON CAUSE FTS	1.005	178

UHS = ultimate heat sink; AEPS = alternate emergency power system; RWST = refueling water storage tank

Note 1 – The current plant procedures and training meet current industry standards. There are no additional specific procedure improvements that could be identified that would affect the result of the HEP calculations. Therefore, no SAMA items were added to the plant specific list of SAMAs as a result of the human actions on the list of basic events with RRW greater than 1.005.

CALLAWAY PLANT UNIT 1
LICENSE RENEWAL APPLICATION

REQUEST FOR ADDITIONAL INFORMATION

Callaway Level 2 Analysis, Appendix E

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SR	CATEGORY_II	ROADMAP
LE-A1	IDENTIFY those physical characteristics at the time of core damage that can influence LERF. Examples include (a) RCS pressure (high RCS pressure can result in high pressure melt ejection) (b) status of emergency core coolant systems (failure in injection can result in a dry cavity and extensive Core Concrete Interaction) (c) status of containment isolation (failure of isolation can result in an unscrubbed release) (d) status of containment heat removal (e) containment integrity (e.g., vented, bypassed, or failed) (f) steam generator pressure and water level (PWRs) (g) status of containment inerting (BWRs)	Section 2.3, Plant Damage States
LE-A2	IDENTIFY the accident sequence characteristics that lead to the physical characteristics identified in LE-A1. Examples include (a) type of initiator (1) Transients can result in high RCS pressure (2) LOCAs usually result in lower RCS pressure (3) ISLOCAs, SGTRs can result in containment bypass (b) status of electric power: loss of electric power can result in loss of ECC injection (c) status of containment safety systems such as sprays, fan coolers, igniters, or venting systems: operability of containment safety systems determines status of containment heat removal. The references in Notes (1) and (2) provide example lists of typical characteristics.	Section 2.2, 2.3, & CET
LE-A3	IDENTIFY how the physical characteristics identified in LE-A1 and the accident sequence characteristics identified in LE-A2 are addressed in the LERF analysis. For example (a) which characteristics are addressed in the Level 1 event trees (b) which characteristics, if any, are addressed in bridge trees (c) which characteristics, if any, are addressed in the containment event trees. JUSTIFY any characteristics identified in LE-A1 or LE-A2 that are excluded from the LERF analysis.	Section 2.2 & CET
LE-A4	PROVIDE a method to explicitly account for the LE-A1 and LE-A2 characteristics and ensure that dependencies between the Level 1 and Level 2 models are properly treated. Examples include: treatment in Level 2, expanding Level 1, construction of a bridge tree, transfer of the information via PDS, or a combination of these.	Integrated model, section 2.3
LE-A5	DEFINE plant damage states consistent with LE-A1, LE-A2, LE-A3, and LE-A4.	Section 2.3
LE-B1	IDENTIFY LERF contributors from the set identified in Table 2-2.8-3. INCLUDE as appropriate, unique plant issues as determined by expert judgment and/or engineering analyses.	Section 2.2

SR	CATEGORY_II	ROADMAP
LE-B2	DETERMINE the containment challenges (e.g., temperature, pressure loads, debris impingement) resulting from contributors identified in LE-B1 using applicable generic or plant-specific analyses for significant containment challenges. USE conservative treatment or a combination of conservative and realistic treatment for non-significant containment challenges. If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	Section 2.2 subsections on Containment Failure at Vessel Breach & Containment Heat Removal Fails
LE-B3	UTILIZE supporting engineering analyses in accordance with the applicable requirements of Table 2-2.3-2(b).	Success criteria based on system notebooks for containment systems, which provide SC basis.
LE-C1	DEVELOP accident sequences to a level of detail to account for the potential contributors identified in LE-B1 and analyzed in LE-B2. Compare the containment challenges analyzed in LE-B with the containment structural capability analyzed in LE-D and identify accident progressions that have the potential for a large early release. JUSTIFY any generic or plant-specific calculations or references used to categorize releases as non-LERF contributors based on release magnitude or timing. NUREG/CR-6595, App. A provides a discussion and examples of LERF source terms.	Section 2.2 subsection on Containment Failure at Vessel Breach, CET, Section 2.4 - Level 2 Sequences, Section 2.5 - Release Categories
LE-C2	INCLUDE realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	Section 2.2 subsections RCS Depressurize Early; Section 3.2.5 – Sensitivity Studies
LE-C3	REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair [i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability (see SY-A24, DA-C15, and DA-D8)]. AC power recovery based on generic data applicable to the plant is acceptable.	Section 3.2.4

SR	CATEGORY_II	ROADMAP
LE-C4	INCLUDE model logic necessary to provide a realistic estimation of the significant accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating staff, effect of fission product scrubbing on radionuclide release, and expected beneficial failures in significant accident progression sequences. PROVIDE technical justification (by plant-specific or applicable generic calculations demonstrating the feasibility of the actions, scrubbing mechanisms, or beneficial failures) supporting the inclusion of any of these features.	Section 2.2, CET, Section 2.4 - Level 2 Sequences, fission product scrubbing in MAAP as appropriate, beneficial failures by e.g., LOCAs that depressurize
LE-C5	USE appropriate realistic generic or plant-specific analyses for system success criteria for the significant accident progression sequences. USE conservative or a combination of conservative and realistic system success criteria for non-risk significant accident progression sequences.	System success criteria in system notebooks
LE-C6	DEVELOP system models that support the accident progression analysis in a manner consistent with the applicable requirements for 2-2.4, as appropriate for the level of detail of the analysis.	Existing system models for CS, VN, CIS
LE-C7	In crediting HFEs that support the accident progression analysis, USE the applicable requirements of 2-2.5, as appropriate for the level of detail of the analysis.	Conservative treatment of HFE, with sensitivity study
LE-C8	INCLUDE accident sequence dependencies in the accident progression sequences in a manner consistent with the applicable requirements of para. 2-2.2, as appropriate for the level of detail of the analysis.	Integrated model, section 2.3
LE-C9	JUSTIFY any credit given for equipment survivability or human actions under adverse environments.	Section 3.2.4
LE-C10	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for nonsignificant accident progression sequences.	Section 3.2.4
LE-C11	JUSTIFY any credit given for equipment survivability or human actions that could be impacted by containment failure.	Section 3.2.4

SR	CATEGORY_II	ROADMAP
LE-C12	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	Section 3.2.4
LE-C13	PERFORM a containment bypass analysis in a realistic manner. JUSTIFY any credit taken for scrubbing (i.e., provide an engineering basis for the decontamination factor used).	Section 2.2 subsection on No Large Early Release, MAAP model for SGTR with scrubbing
LE-D1	DETERMINE the containment ultimate capacity for the containment challenges that result in a large early release. PERFORM a realistic containment capacity analysis for the significant containment challenges. USE a conservative or a combination of conservative and realistic evaluation of containment capacity for non-significant containment challenges. If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated. Analyses may consider use of similar containment designs or estimating containment capacity based on design pressure and a realistic multiplier relating containment design pressure and median ultimate failure pressure. Quasi-static containment capability evaluations are acceptable unless hydrogen concentrations are expected to result in potential detonations. Such considerations need to be included for small volume containments, such as the ice condenser type.	Section 2.2 subsections on Containment Failure Early & Containment Heat Removal, based on plant-specific values of WCAP
LE-D2	EVALUATE the impact of containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows and INCLUDE as potential containment challenges, as required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	Covered by IPE Section 4.1.1, Containment Structure and Systems, subsection Containment Isolation
LE-D3	When containment failure location affects the event classification of the accident progression as a large early release, DEFINE failure location based on a realistic containment assessment that accounts for plant-specific features. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	NA - Failure location does not affect classification of early failures

SR	CATEGORY_II	ROADMAP
LE-D4	PERFORM a realistic interfacing system failure probability analysis for the significant accident progression sequences resulting in a large early release. USE a conservative or a combination of conservative and realistic evaluation of interfacing system failure probability for nonsignificant accident progression sequences resulting in a large early release. INCLUDE behavior of piping relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions.	See ISLOCA References
LE-D5	PERFORM a realistic secondary side isolation capability analysis for the significant accident progression sequences caused by SG tube failure resulting in a large early release. USE a conservative or a combination of conservative and realistic evaluation of secondary side isolation capability for nonsignificant accident progression sequences resulting in a large early release. JUSTIFY applicability to the plant being evaluated. Analyses may consider realistic comparison with similar isolation capability in similar containment designs.	Section 2.2, 2.3, CET realistically address SGTRs, binning into appropriate PDSs, sensitivities for scrubbing and late releases
LE-D6	PERFORM an analysis of thermally-induced SG tube rupture that includes plant-specific procedures and design features and conditions that could impact tube failure. An acceptable approach is one that arrives at plant-specific split fractions by selecting the SG tube conditional failure probabilities based on NUREG-1570 or similar evaluation for induced SG failure of similarly designed SGs and loop piping. SELECT failure probabilities based on (a) RCS and SG post-accident conditions sufficient to describe the important risk outcomes (b) secondary side conditions including plant-specific treatment of MSSV and ADV failures. JUSTIFY assumptions and selection of key inputs. An acceptable justification can be obtained by the extrapolation of the information in NUREG-1570 to obtain plant-specific models, use of reasonably bounding assumptions, or performance of sensitivity studies indicating low sensitivity to changes in the range in question.	Appendix D
LE-D7	PERFORM containment isolation analysis in a realistic manner for the significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative or realistic treatment for the non-significant accident progression sequences resulting in a large early release. INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions.	See CIS system notebook / IPE

SR	CATEGORY_II	ROADMAP
LE-E1	SELECT parameter values for equipment and operator response in the accident progression analysis in a manner consistent with the applicable requirements of 2-2.5 and 2-2.6 including consideration of the severe accident plant conditions, as appropriate for the level of detail of the analysis.	Equipment and operator parameters consistent with Level 1 approach
LE-E2	USE realistic parameter estimates to characterize accident progression phenomena for significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative and realistic estimates for non-significant accident progression sequences resulting in a large early release.	Phenomena values based on plant-specific values, WCAP
LE-E3	INCLUDE as LERF contributors potential large early release (LER) sequences identified from the results of the accident progression analysis of LE-C except those LER sequences justified as non-LERF contributors in LE-C1.	Section 2.5
LE-E4	QUANTIFY LERF in a manner consistent with the applicable requirements of Tables 2-2.7-2(a), 2-2.7-2(b), and 2-2.7-2(c).	Section 3.2
LE-F1	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 2-2.8-3.	Section 3.2
LE-F2	REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant specificity is appropriate for significant contributors, etc.).	Section 3.2.4
LE-F3	IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, in a manner consistent with the applicable requirements of Tables 2-2.7-2(d) and 2-2.7-2(e).	Section 3.2
LE-G1	DOCUMENT the LERF analysis in a manner that facilitates PRA applications, upgrades, and peer review.	Entire Level 2 Report

SR	CATEGORY_II	ROADMAP
LE-G2	DOCUMENT the process used to identify plant damage states and accident progression contributors, define accident progression sequences, evaluate accident progression analyses of containment capability, and quantify and review the LERF results. For example, this documentation typically includes (a) the plant damage states and their attributes, as used in the analysis (b) the method used to bin the accident sequences into plant damage states (c) the containment failure modes, phenomena, equipment failures and human actions considered in the development of the accident progression sequences and the justification for their inclusion or exclusion from the accident progression analysis (d) the treatment of factors influencing containment challenges and containment capability, as appropriate for the level of detail of the analysis (e) the basis for the containment capacity analysis including the identification of containment failure location(s), if applicable (f) the accident progression analysis sequences considered in the containment event trees (g) the basis for parameter estimates (h) the model integration process including the results of the quantification including uncertainty and sensitivity analyses, as appropriate for the level of detail of the analysis.	Section 2
LE-G3	DOCUMENT the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.	Section 3.2
LE-G4	DOCUMENT the sources of model uncertainty and related assumptions (as identified in LE-F3) associated with the LERF analysis, including results and important insights from sensitivity studies.	Assumptions in Section 2.1; Key uncertainties in Section 3.2.5
LE-G5	IDENTIFY limitations in the LERF analysis that would impact applications.	Section 3.2.1
LE-G6	DOCUMENT the quantitative definition used for significant accident progression sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	Section 3.2.4