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September 17, 2010
GO2-10-138

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION LICENSE RENEWAL APPLICATION**

References:
1) Letter, G02-10-011, dated January 19, 2010, WS Oxenford (Energy Northwest) to NRC, "License Renewal Application"

2) Letter dated July 13, 2010, NRC to WS Oxenford (Energy Northwest), "Request for Additional Information for the Review of the Columbia Generating Station License Renewal Application – SAMA Review," (ADAMS Accession No. ML101760421)

Dear Sir or Madam:

By Reference 1, Energy Northwest requested the renewal of the Columbia Generating Station (CGS) operating license. Via Reference 2, the Nuclear Regulatory Commission (NRC) requested additional information related to the Energy Northwest submittal.

Transmitted herewith in Attachment 1 is the Energy Northwest response to a Request for Additional Information (RAI) contained in Reference 2. Enclosure 1 contains Amendment 10 to the License Renewal Application (LRA) that was submitted in Reference 1.

Certain parts of the RAIs in this response letter will be addressed later by providing a sensitivity study, due to the NRC by October 15, 2010.

No new commitments are included in this response.

If you have any questions or require additional information, please contact Abbas Mostala at (509) 377-4197.

A035
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on the date of this letter.

Respectfully,

9/17/10

SK Gambhir
Vice President, Technical Services

Attachment 1: Response to Request for Additional Information

Enclosure 1: Amendment 10 to the LRA

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
EJ Leeds - NRC NRR
EFSEC Manager
RN Sherman – BPA/1399
WA Horin – Winston & Strawn
D Doyle - NRC NRR (w/a)
BE Holian - NRC NRR
RR Cowley – WD
E Gettys (w/a)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

NRC Request:

1) Provide the following information regarding the Level 1 Probabilistic Safety Assessment (PSA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

   a. Environmental Report (ER) Section E.3 explains that the SAMA evaluation is based on PSA Revision 6.2 models for Level 1 and 2 internal, fire, and seismic events. Table E.3-1 shows the completion dates for these different models to range from 1/2004 to 2/2007 and shows these models to be based on incorporation of plant modifications that occurred up through 8/2006 and Columbia Generating Station (CGS) data/Bayesian updating through 6/2002. Identify any changes to the plant (physical and procedural modifications) since those dates that could have a significant impact on the results the SAMA analyses, and provide a qualitative assessment of their impact on the PSA and on the results of the SAMA evaluation. Identify whether a newer PSA model is available, and if so, provide a brief description of the major changes relative to the PSA Revision 6.2, and provide an assessment of the impact on the results of the SAMA evaluation (e.g., increased benefit or additional SAMAs if the baseline core damage frequency (CDF) has increased; any new candidate SAMAs for newly-identified dominant sequences or risk-significant basic events).

Energy Northwest Response to 1.a:

The permanent plant physical changes that have occurred that potentially could alter the results of the SAMA analysis presented in Table E.11-1 are:

1. The ability to cross-connect diesel generator (DG) 3 to either the Division 1 or Division 2 emergency buses during an extended station blackout (SBO) was implemented along with a significant change to the loss of offsite power (LOOP) and SBO procedures. This additional feature was in response to the DG Completion Time Technical Specification (TS) change as a risk management action but not credited in the PSA. This addition to the plant significantly improves the plant's ability to cope with a LOOP and an SBO. SAMA cost benefit cases associated with many of the AC/DC cases that would enhance plant response to a LOOP or SBO become less beneficial when this plant change is incorporated in the PSA; therefore, the SAMA analysis is conservative relative to this modification.

2. DG-4, a portable 480 V DG, was added, along with significant procedure changes for its use, to provide an alternate source of AC power to charge the emergency batteries for Division 1 and Division 2 and the battery that provides

breaker control power to recover offsite power from the 230 kV grid. The current licensing basis is to have DG-4 available whenever a DG is in an extended outage (greater than 72 hours). This modification improves the ability to maintain reactor core isolation cooling (RCIC) operating by extending the battery life if an SBO should occur during an online DG outage. This additional feature was added to address a risk management action for the DG Completion Time TS change but had not been credited in the PSA. Although this change improves the plant's ability to cope with an SBO with one DG inoperable, SAMA candidate AC/DC-03 was proposed to look at the potential benefit that would occur by staging DG-4 permanently in a location that made it available continuously while operating. The SAMA evaluation therefore explicitly addresses this modification.

3. An upgrade of the feedwater control system and the turbine control system was implemented. Credit for anticipated higher reliability of these features was not incorporated into the Rev. 6.2 PSA. Past experience indicated that a period of use was necessary to ensure an actual reliability improvement would result. The periodic update of the plant equipment and event failure history by Bayesian update has not occurred for these two plant improvements. Sufficient operational time is necessary prior to capturing that history within the PSA. These changes are being judged as risk neutral at this stage. No SAMA candidates associated with the feedwater and turbine control system were proposed for further cost benefit consideration. See the response to RAI 5.a regarding a sensitivity will be performed for manual control of feedwater flow without DC power.

4. Other than indicated above, no procedural changes that affect operation or maintenance of the plant have been identified that would significantly impact the SAMA results. However, a potential SAMA candidate for enhancements to procedures and training is addressed in the response to RAI 5.e.

Thus, the four changes above would generally result in a reduction in the SAMA candidate's risk reduction worth (RRW) and reduce the benefit provided in Table E.11-1.

A newer Internal Events PSA Level 1 and Level 2 model, Rev. 7.1, is now available that has been upgraded to Regulatory Guide (RG) 1.200 Rev. 2 "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" through the peer review process. This model was not available at the time the SAMA analysis was performed. Modeling enhancements made as a result of upgrading the PSA to meet RG 1.200 Rev. 2 resulted in a higher baseline core damage frequency (CDF) and a lower large early release frequency (LERF). The Fire PSA (FPSA) and Seismic PSA (SPSA) models have not been upgraded, but have been integrated with the new internal events model. This model resolves all facts and observations (F&Os) generated from the reviews listed in Section E.5.2 and will be used to perform sensitivity analyses on the significance of these outstanding PSA changes. These sensitivity evaluations will compare the

Rev. 7.1 delta-CDF and delta-LERF values to those for Rev. 6.2 to identify any impact to the SAMA analysis, identify any new significant sequences, and identify new risk-significant basic events. This will serve to identify any additional SAMA candidates that should be considered. Both the existing and potentially additional SAMA candidates will be assessed, and if significant CDF or LERF benefit is identified, the item will be considered for further cost benefit evaluation. This analysis (also known as a sensitivity study) will be provided to the NRC under separate letter.

NRC Request:

b. ER Table E.3-3 shows the contribution of specific types of Transients and loss of coolant accidents (LOCAs) to core damage frequency (CDF) and identifies specific initiators leading to anticipated transient without scram (ATWS). In light of the fact that station blackout (SBO) contributes 33.1% to the CDF, identify the initiators that contribute to SBO.

Energy Northwest Response to 1.b:

The initiating event contributors to an SBO are the LOOP (unavailability of 230 kV power through the startup transformer (TR-S) via bus ducting and intermediate switchgear and 115 kV power through the backup transformer (TR-B) via electrical cabling to the emergency Division 1 and Division 2 switchgear) and the loss of DG-1 and DG-2. The power output from CGS is to the 500 kV Bonneville Power Administration grid network. CGS electrical design would not result in a turbine trip upon loss of the credited offsite power sources (i.e., CGS is classified as a no-trip LOOP plant). However, the CGS PSA is modeled with the assumption that the turbine generator is tripped on a LOOP. Following a LOOP and the assumption that the main turbine is tripped, the failure of the DGs for Division 1 and Division 2 starts the progression of the SBO initiating event. The CGS PSA model then establishes event trees to model the progression of the accident sequences to core damage.

The LOOP contribution is taken from industry data sources (e.g., NUREG/CR-5496 "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980-1996") coupled with a plant specific evaluation. The three contributors within the LOOP initiator are plant-centered, grid, and weather related contributors based on industry data for the period from 1980 to 1996.

SAMA candidates were evaluated for each of these contributors. See enhancements related to AC and DC power AC/DC-01 through AC/DC-03, AC/DC-10 through AC/DC-12, AC/DC-15 through AC/DC-20, AC/DC-23 through AC/DC-29 and enhancements related to core cooling CC-01, CC-02, CC-03a, CC-03b, CC-05, and CC-07 in Table E.10-1. Many of these are generic candidates and were already implemented at CGS.

Additionally, NUREG/CR-6890 "Reevaluation of Station Blackout Risk at Nuclear Power Plants Analysis of Loss of Offsite Power Events: 1986-2004" Table D-1 lists CGS-specific LOOP frequencies that can be compared to the frequencies used for PSA Rev. 6.2. NUREG/CR-6890 covered the industry LOOP experience from 1986 through 2004. Table 1.b-1 provides a comparison that demonstrates that the LOOP value used for the SAMA cases is conservative relative to current industry data.

The DG failure probabilities were derived from Bayesian update of generic industry failure rate with plant specific failure experience.

Table 1.b-1: Loop Frequencies for CGS at Power Compared to NUREG/CR-6890

LOOP Category	LOOP Frequency for SAMA Analysis	LOOP Frequency in NUREG/CR-6890 Table D-1
Plant-centered (/ Reactor-Critical-Year)	4.38E-02	2.01E-03
Switchyard (/ Reactor-Critical-Year)	Included in plant-centered	9.08E-03
Grid-related (/ Reactor-Critical-Year)	4.00E-03	1.49E-02
Severe-weather (/ Reactor-Critical-Year)	5.30E-03	3.85E-03
Total (/ Reactor-Critical-Year)	5.31E-02	2.98E-02
Total (/Reactor-Year)	3.6E-02	2.5E-02

The Rev. 6.2 PSA does not include modeling for consequential LOOP (plant trip causing LOOP) as a contributor for SBO. However, the data from NUREG/CR-6890 includes this consequential LOOP industry experience in deriving the LOOP frequencies.

Conclusion: The initiators that contribute to the SBO initiating event frequency are initiating event frequency for loss of CGS 230 kV and 115 kV (assuming turbine trip) combined with a loss of the Division 1 and Division 2 onsite emergency DGs.

A comparison of results of the LOOP initiation frequency of the current model of 5.31E-02/reactor critical-year to the NUREG/CR-6890 LOOP initiation frequency of 2.98E-02/reactor critical-year, which includes consequential LOOP, demonstrates that the contribution to the SAMA candidates would not be impacted by later industry LOOP experience or consequential LOOP.

NRC Request:

c. ER Section E.5.2 presents a list of seven technical reviews (covering internal events and fire, and Level 1 and Level 2) of the PSA (page E-31) and a list of four external peer reviews (page E-32) that contributed to updating the PSA models. Provide the following relative to these reviews:

   i) A summary of the scope of the 1997 owner's group peer review mentioned in Section E.5.5.

   ii) A brief description of all unresolved B Level facts and observations (F&Os) from the 2004 internal events PSA peer review discussed in Section E.5.5 and an assessment of their impact on the SAMA evaluation.

   iii) A summary of the scope of the other two peer reviews and the seven technical reviews discussed in Section E.5.2, a brief description of all unresolved issues/F&Os, and an assessment of their impact on the SAMA evaluation.

Energy Northwest Response to 1.c:

i) The 1997 CGS peer review was performed on Revision 3 of the Internal Events PSA. The revision included all technical elements, including internal flooding, for the Level 1 and Level 2 internal events model and documentation. The scope of the review included the following:

Level 1 PSA Certification Elements

- Initiating Events (IE)
  - Guidance documents for initiating event analysis
  - Groupings
    - Transients
    - LOCA
    - Support system/special initiators
    - Inter-System LOCA (ISLOCA)
    - Break outside containment (BOC)
    - Internal floods
  - Subsumed Events
  - Data
  - Documentation
- Accident Sequence Evaluation
  - Guidance on development of event trees
  - Event trees (accident scenario evaluation)
    - Transients
    - SBO

            - LOCA
            - ATWS
            - Special initiators
            - ISLOCA & BOC
            - Internal floods
        - Documentation
- Thermal Hydraulic Analysis
    - Guidance Document
    - Best estimate calculations [e.g., Modular Accident Analysis Program (MAAP)]
    - Generic assessments
    - FSAR - Chapter 15
    - Room heat up calculations
    - Documentation
- System Analysis (SY) including fault trees
    - System analysis guidance documents
    - System models
        - Structure of models
        - Level of detail
        - Success criteria
        - Nomenclature
        - Data
        - Dependencies
        - Assumptions
    - Documentation of system notebooks
- Data Analysis
    - Guidance
    - Component failure probabilities
    - Systems/trains maintenance unavailability
    - Unique unavailability or modeling items
        - AC recovery
        - Scram system
        - Emergency DG mission time
        - Repair and recovery model
        - Stuck open relief valve (SORV)
        - LOOP given transient
        - Balance of plant (BOP) unavailability
        - Pipe rupture failure probability
    - Documentation
- Human Reliability Analysis (HRA)
    - Guidance
    - Pre-initiator human actions
        - Identification
        - Analysis
        - Quantification

    - Post-initiator human actions and recovery
        - Identification
        - Analysis
        - Quantification
    - Dependence among actions
    - Documentation
- Dependencies
    - Guidance document on dependency treatment
    - Intersystem dependencies
    - Treatment of human interactions
    - Treatment of common cause
    - Treatment of spatial dependencies
    - Walkdown results
    - Documentation
- Structural Capability
    - Guidance
    - Reactor pressure vessel (RPV) capability (pressure and temperature)
        - ATWS, transient
    - Containment (pressure and temperature)
    - Reactor building
    - Pipe over-pressurization for ISLOCA
    - Documentation
- Quantification Results Interpretation
    - Guidance
    - Computer Code
    - Simplified model (e.g. cutset model usage)
    - Dominant sequences/cutsets
    - Recovery analysis
    - Truncation
    - Uncertainty
    - Results summary

<u>Level 2 PSA Certification Elements</u>

- Guidance document
- Success criteria
- L1/L2 interface
- Phenomena considered
- Important human error probabilities (HEPs)
- Containment capability assessment
- End state definition
- LERF definition
- Containment event trees (CETs)
- Documentation

PSA Maintenance and Update Process Certification

- Maintenance and Update Process
  - Guidance document
  - Input-monitoring and collecting new information
  - Model control
  - PSA maintenance and update process
  - Evaluation of results
  - Re-evaluation of past PSA applications
  - Documentation

ii) The peer review performed in 2004 was on the Revision 5.0 Internal Events Level 1 and Level 2 PSAs against the American Society of Mechanical Engineers (ASME) Standard RA-Sa-2003 "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" and RG 1.200 For Trial Use. The PSA's ability to meet the Supporting Requirements (SR) were assessed against the Capability grading criteria of the ASME standard and RG 1.200 (Capability Category I, II, III or not met). Additionally, since a FPSA standard was unavailable, a review was performed on the FPSA to the high level requirements identified in the ASME standard.

Table 1.c.ii-1 lists the unresolved internal events Level B F&Os from the 2004 peer review that potentially could have an impact on the model results and have not been incorporated in PSA Revision 6.2, and an assessment of their impact on the SAMA evaluation. Additionally, the peer review team's capability grading is included. F&Os classified as B level that were documentation-related only are not included. All but two of the Supporting Requirements were met, at least at Capability Category I.

Table 1.c.ii-2 identifies one unresolved 2004 peer review finding that would have been graded as not met to the ASME standard high level requirements for the FPSA. Table 1.c.ii-2 provides an assessment of its impact on the SAMA evaluation.

It is our judgment that these items won't significantly impact the SAMA analysis findings. Some of the unresolved F&Os address conservatisms and thus are conservative relative to the SAMA evaluation.

Additionally, the majority of the unresolved F&Os were related to the LERF results. The sensitivity studies performed in Section E.7 demonstrate that there is not a high sensitivity to changes in LERF due to the low population within the evacuation zone. Further, the significant F&Os have been resolved and incorporated in the upgraded model and the individual and cumulative effects can be assessed. Further details are provided in the tables.

Table 1.c.ii-1: Significant (B level) Unresolved Internal Events Peer Review F&Os for Impacts to SAMA Evaluation				
F&O	SR[1]	Observations	Recommendations	Resolution
IF-E5-1	IF-C6	The following have been identified as potential areas for improvement in the internal flooding HRA: 1) The isolation HEP could vary depending on the type and location of the break. The HRA documentation/quantification does not seem to address how or if these factors were accounted for in the quantification. The type of factors that could impact the quantification might include procedure availability/nature of directions, cues, indication availability, travel considerations, special equipment requirements (keys, lights, ladders), and what isolation equipment is available. 2) The manipulation time assumed for flood isolation is "generally assumed" to be 30 minutes; however, it appears that actions in the 20 minute time frame are credited (should be 1.0 if the manipulation time exceeds the available time). Clarification of the timing for flooding scenarios would be beneficial. 3) The documentation does not clearly identify how each component of the HEP was derived. While the case identifier of "Step by step, Extremely High Stress in the Lower Bound" is provided, a specific reference for each failure probability would	Re-perform the flooding HEPs accounting for the factors identified above. Confirm that the method used is consistent with the current internal events HRA and that the quantification is performed with the same level of detail. If resources permit, the use of more recent HRA methods such as the Cause Based Method augmented by a time reliability correlation is considered desirable.	Peer review found that SR meets Capability III with comments to address conservatisms.[2] The flooding HEPs are judged to be conservative. Resolution of this F&O would reduce model conservatism associated with flooding HEPs.

Table 1.c.ii-1: Significant (B level) Unresolved Internal Events Peer Review F&Os for Impacts to SAMA Evaluation				
F&O	SR[1]	Observations	Recommendations	Resolution
		be helpful.		
IF-E7-1 LE-A4-1	IF-E7 LE-A4	LERF Impact: The flood evaluation transfers core damage end states from the Level 1 into Level 2. These core damage end states are assigned in a manner similar to other Level 1 event trees. In addition, the flood scenarios assign one of the sequences, where no flood isolation occurs, to a new class, Class IC. This Class is transferred to Level 2 where all of this frequency (IC) is treated as LERF. No physical description of the cutsets in this transferred plant damage state (PDS), IC, is provided that would justify its use as a LERF contributor. In general, these events are believed to be low pressure core damage cases or long term Loss of decay heat removal (DHR) cases, either of which is non-LERF.	Reconsider the treatment of these flood sequences in Level 2 to reduce excess conservatisms in the LERF assessment.	Peer review found that the SR meets Capability I.[2] Resolution of this F&O will reduce model conservatism associated with PDS Class C and its release categorization. This would reduce the conservatism in the LERF component benefit relative to the input to the SAMA evaluation, making it more likely that SAMA candidates

Table 1.c.ii-1: Significant (B level) Unresolved Internal Events Peer Review F&Os for Impacts to SAMA Evaluation				
F&O	SR[1]	Observations	Recommendations	Resolution
				would appear cost beneficial.
LE-A4-1	LE-A4	The Level 2 analysis appears to be consistent with NUREG/CR-6595 "An Approach for Estimating the Frequencies of Various Containment Bypass Failure Modes and Bypass Events," which the NRC has identified as acceptable for Capability Category I. One of the reasons for limiting the Capability Category is that dependencies are difficult to transfer into Level 2 for accurate treatment. Some of these dependencies are related to: 1) Special initiators that may adversely impact parts of systems and 2) Internal floods that may adversely impact parts of systems or be treated in an excessively conservative manner	Increasing the capability category for Level 2 would require a more rigorous treatment of dependencies from Level 1 to Level 2.	Peer review found that SR meets Capability I.[2]
LE-B2-1 LE-D1-1	LE-B2 LE-D1a	The ex-vessel steam explosion is quantitatively assessed in the PSA model for its potential LERF impact. The assessment references several small scale PWR tests. It is judged that this may result in some conservative treatment of the ex-vessel steam explosion for the Mark II BWR. In addition, there are accident sequences in Level 2 in which there is no water available in the CGS recessed cavity	If resources permit, removal of excess conservatism in the ex-vessel steam explosion quantitative impact would be desirable.	Peer review found that SRs meet Capability II.[2] Resolution of this F&O will reduce model conservatism, which is

Table 1.c.ii-1: Significant (B level) Unresolved Internal Events Peer Review F&Os for Impacts to SAMA Evaluation				
F&O	SR[1]	Observations	Recommendations	Resolution
		to support a steam explosion. Under such conditions, the assumption of an ex-vessel steam explosion leads to a conservative characterization of LERF. This occurs in some short term SBO accident sequences as well as other transient induced containment challenges.		conservative relative to the SAMA evaluation.
LE-C2-1	LE-C2a	Crew Actions: The crew actions included in the LERF assessment are 1) not explicitly tied to procedural direction, 2) do not account for failures that have previously occurred in the Level 1, 3) are all assessed individually to be 0.1 regardless of dependency issues.	The HRA interface in Level 2 with phenomena and dependencies in Level 1 can be quite complex. The simplistic treatment of assigning 0.1 estimates to HEPs without accounting for dependencies would appear to need additional justification. Examples are high pressure core melt sequences from Level 1 with crew failure to depressurize, which are afforded additional "credit" for RPV depressurization in Level 2 using the assumed 0.1 HEP with no account for dependencies.	Peer review found that SR meets Capability I.[2]
LE-C3-1	LE-C3	Survivability of Systems: PSA-2-L2-001 (Rev 2 Model Rev 5) p. 69 states that some general conservative assumptions were made for Level 2 such as 2 Train System = 0.05 and 1 Train System = 0.1. In practice, the CGS PSA model in some cases may multiply (0.1) for each "available" train. These assumptions may not always be	If Capability Category II is required, then these assumptions need to be replaced with specific analysis to address equipment survivability.	Peer review found that SR meets Capability I.[2]

Table 1.c.ii-1: Significant (B level) Unresolved Internal Events Peer Review F&Os for Impacts to SAMA Evaluation				
F&O	SR[1]	Observations	Recommendations	Resolution
		conservative because of directly induced failures (e.g., containment failure). In addition, the use of "conservative assumptions" always leads to the assignment of Capability Category I.		
LE-C4/C5-1	LE-C4	System Success Criteria: The assessment of system success is based on the assignment of arbitrary conditional failure probabilities. No detailed evaluation of the system success is performed. This is generally adequate for a NUREG/CR-6595 type of Level 2/LERF analysis.	Improve Level 2 analysis to explicitly model the systems credited in Level 2 with fault tree models and the dependency treatment relative to Level 1 sequence cutsets and phenomena occurring after core damage.	Peer review found that SR meets Capability I.[2]
LE-C4-2	LE-C4	Vent Sequences and Induced Core Damage: The declaration of a General Emergency occurs at the time of vent declaration in the PSA. However, vent sequence successes are not treated as leading to core damage. Depending on the specific operator training and procedures, the technique for venting may create a rapid blowdown of containment and steam binding of pumps taking suction from the suppression pool.	Re-examine the assumptions regarding containment venting and the ability to preserve adequate core cooling following venting. If there is confirmed to be no vent impact on adequate core cooling capability, then no further effort is required. If there is an impact, then the correct Emergency Action Level and EOP directions need to be factored into the assessment of potential releases to determine whether they represent LERF releases. Define a MAAP case that shows timing of a release (i.e., time to core damage after vent) such that it would not be a LERF. It is not expected that vent sequences will result in LERF.	Peer review found that SR meets Capability I.[2]

Table 1.c.ii-1: Significant (B level) Unresolved Internal Events Peer Review F&Os for Impacts to SAMA Evaluation				
F&O	SR[1]	Observations	Recommendations	Resolution
LE-C7-2	LE-C7	Accident Sequence Dependencies: Transfers of dependencies from Level 1 are not performed except at the PDS level.	If Capability Category II is required, improve the Level 1 to Level 2 transfer of accident sequences.	Peer review found that SR meets Capability I.[2]
LE-F1-1	LE-F1a	LERF Summary: The LERF summary is considered adequate for applications. The computer code is capable of developing additional quantitative characterizations of significant contributors.	Provide the specific LERF contributor assessment requested in the ASME PSA Standard and review it for reasonableness.	Peer review found that SR is not met. This is primarily a documentation F&O but limits the ability to identify basic event LERF contributors.[2]
SC-B1-4	SC-B1	Room Cooling: The switchgear room heat-up calculations should be performed to eliminate conservative biases in the results due to room cooler requirements.	Complete the on-going switchgear room cooling calculation and incorporate results into the analysis. The current modeling uses modeling that is judged to be reasonably conservative.	Peer review found that SR is not met. Recent calculation confirmed the PSA Rev. 6.2 modeling is conservative.[2]

1. The applicable ASME Standard Addendum B supporting requirement.

2. This F&O was resolved in the upgraded Internal Events PSA. The sensitivity study identified in the response to RAI 1.a will identify any collective impact to the SAMA analysis.

Table 1.c.ii-2: Review of Significant Unresolved FPSA Peer Review F&Os for Impacts to SAMA Evaluation				
F&O	SR	Observations	Recommendations	Resolution
FS-B4/B5-1	N/A	The CGS FPSA conservatively does not credit fire brigade response (this fact is appropriately noted in the CGS FPSA documentation). It may be appropriate to consider credit for fire brigade response in future updates.	Consider credit for fire brigade response in future updates. Caution should be used for the crediting of fire brigades to ensure that double credits are not taken (i.e., ensure that fire brigade activities are not inherently credited in other sequence nodes such as "early extinguishments" – it is believed that CGS already ensures this particular aspect).	The peer review found the high level criteria for FPSA was not met. Resolution of this F&O will reduce model conservatism, which is conservative relative to the SAMA evaluation.

iii. A summary of the scope of the other two reviews (NRC Inspection and Mitigating Systems Performance Index (MSPI) Inspection) and the seven technical reviews discussed in Section E.5.2 are provided below. All significant unresolved issues/F&Os that would impact the quantitative result from these inspections and an assessment of their impact are provided in Tables 1.c.ii-1 and 1.c.ii-2 above.

Other Peer Review 1, 2004 NRC Inspection to RG 1.200 Pilot Plant Program: The NRC inspection was a multi-member team with expertise in all elements of the PSA to review the CGS Internal Events Level 1 and Level 2 PSA Revision 5.0. The reviewers used the ASME standard RA-Sa-2002 Addendum A and RG 1.200 for trial use and assessed the CGS Internal Events Level 1 and Level 2 PSA to each element and supporting requirement. The team's approach for the review followed a very similar pattern that the 2004 peer review team used. The NRC was provided a copy of the 2004 peer review team's report and associated F&Os as a reference, but conducted the review and made their own independent assessment of the technical adequacy of the CGS PSA to meet the individual supporting requirements. The inspection also focused on the usability of the ASME standard and the RG by a peer review team to assess the technical adequacy of the PSA. The NRC review team's issues were treated using the same process as peer review team F&Os by Energy Northwest and documented, resolved, incorporated, or deferred as appropriate based on their risk significance. Unresolved issues were reported in the Appendix E submittal. The sensitivity study identified in the response to RAI 1.a will collectively address any significant unresolved issues and their impact on the SAMA analysis.

Other Peer Review 2, 2006 MSPI Inspection: The inspectors completed an inspection in accordance with Temporary Instruction 2515/169 "Mitigating Systems Performance Index Verification" to verify that CGS correctly implemented the MSPI guidance. The inspectors reviewed the basis document for significant issues that resulted in a change to the system boundary, an addition of a monitored component, or a change in the reported index color. The inspectors reviewed the data CGS used to generate the MSPI basis document and actual unavailability and unreliability values. The inspectors also used the following CGS source documents to verify the validity of the input data:

- Control Room Logs
- Surveillance Test Procedures
- Maintenance Procedures

The scope of the MSPI inspection included sampling a portion of the MSPI systems for review and verifying actual data input for unavailability and unreliability for these systems.

There were no unresolved issues from this NRC inspection.

1994 Technical Review 1: This review was performed by an independent consultant on the Revision 0 and Revision 1 Individual Plant Examination (IPE). The WNP-2 IPE Report, Revision 0, was reviewed by the in-house review team and by IPEP (Tenera, L.P. and Fauske and Associates, Inc.) Revision 1 of the report was reviewed by NUS Corporation and Tenera, L.P. These reviews were conducted at various stages of report completeness to ensure resolution of comments and to address technical correctness and issues of documentation content and clarity. The review comments were evaluated, and in all cases, satisfactory resolution was achieved between the in-house review team and the reviewers. Subsequently in 1997, the Boiling Water Reactor Owner's Group (BWROG) peer review was conducted.

2004 Technical Review 2, Mechanism Operated Cell (MOC) Switch Assemblies: The MOC switch review was performed to assess a common cause condition associated with the 4160 V switchgear that impacted emergency core cooling system (ECCS) pump motor breakers and other critical distribution breakers. This was a special review focused on the AC distribution system. Modeling changes were implemented to address this failure mode. All identified issues were resolved.

2002-2003 Technical Review 3: A review and upgrade of the Internal Events PSA Revision 4.2 was performed by Scientech to support the DG Completion Time TS change. The process did not produce a formal report, but did establish a two-year project plan for addressing the identified issues and resolving them. This effort was augmented by supporting an NRC request for a RG 1.200 pilot plant that would submit a type of licensing application that had been previously approved by the NRC for other plants. The timing of the Energy Northwest submittal was appropriate for this application to be reviewed using RG 1.200 Trial Use. See above discussion on 2004 peer review and NRC inspection (Other Peer Review 1).

2002 Technical Review 4, Self Assessment for SY and IE elements: To augment the review and upgrade effort for the 2002-2004 revision of PSA 4.2, an internal technical self assessment was performed by Energy Northwest staff. This review identified changes to the SY and IE elements of the Revision 4.2 PSA that were implemented in the upgrade and subsequently evaluated by the 2004 peer review and NRC inspection of the Internal Events PSA Revision 5.0.

2004 Technical Review 5: In response to the peer review of Revision 5.0, a HRA finding associated with SBO initiating events was identified. A focused HRA review was performed, with subsequent upgrade of the PSA. These results were also used to respond to the HRA-related RAIs associated with the DG Completion Time TS change. This review identified a number of additional human failure events for inclusion in the PSA. Revision 5.1 resolved these HRA issues. Unresolved issues were reported in Section E.5.5.1. The sensitivity study identified in response to RAI 1.a will collectively address any significant unresolved issues and their impact on the SAMA analysis.

2006 Technical Review 6, MSPI Support Requirements Self Assessment: This self assessment was performed to assure the PSA Revision 6.0 would meet the implementation requirements for MSPI. There were no Level A F&Os identified by the 2004 peer review for internal events. All 45 Level B F&Os associated with the MSPI application were resolved and incorporated into the MSPI calculations and results. Unresolved issues identified in this technical review, not related to MSPI, were reported in Section E.5.5.1. The sensitivity study identified in response to RAI 1.a will collectively address any significant unresolved issues and their impact on the SAMA analysis.

2008 Technical Review 7: A self assessment was performed of CGS PSA technical adequacy relative to the ASME standard supporting requirements pertinent to a proposed change to the Technical Specification 3.5.1 for extending the completion for the low pressure coolant injection (LPCI) C and low pressure core spray (LPCS) ECCS. This self assessment examined the 2004 CGS peer review F&Os and assessed compliance with Addendum B of the ASME PSA standard as clarified by RG 1.200 Revision 1. Although a TS change has not been submitted for NRC approval (due to reasons other than the risk evaluation), the self assessment was used to identify any unresolved issues. Unresolved issues were reported in Section E.5.5.1. The sensitivity study identified in response to RAI 1.a will collectively address any significant unresolved issues and their impact on the SAMA analysis.

NRC Request:

d. Describe the quality control process for the PSA, including the process of monitoring potential plant changes, tracking items that may lead to model changes, making model changes (including frequency for model updates), documenting changes, software quality control, independent reviews, and qualification of PSA staff.

Energy Northwest Response to 1.d:

The process for controlling the technical adequacy of the PSA for development of the PSA used in the SAMA analysis is contained in engineering procedure SYS-4-34 "PSA Configuration Control". A summary of this procedure associated with PSA maintenance and upgrade is provided below.

In RG 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", the NRC staff expects the PSA to be subjected to quality control when the PSA is used to provide insights into decision making. RG 1.174 describes the pertinent quality control requirements. Consistent with the RG 1.174 guidance, the PSA configuration control procedure ensures that the PSA realistically reflects the actual design, construction, operational practice, and operational experience of the plant. Where the term "should" is used below it denotes recommendations and management expectations.

The PSA configuration control consists of the following tasks:

- Monitor PSA input and collect new information for incorporation into the PSA update. Approved design changes and implementation notices are routed to PSA for review for inclusion in the PSA. Additionally, plant procedure change notices are sent to PSA.

- Update the PSA model to be consistent with the as-built and as-operated plant.

- Assess cumulative impact of pending PSA changes before incorporating into PSA update.

- Control the computer codes used to support PSA quantification in accordance with the software quality assurance program.

1. PSA Inputs and New Information - A review of the PSA input and new information should be conducted once every two years. The following information that could affect the PSA elements, including model, data, methodology, or documents, should be reviewed:

    - Changes in system operating procedures and abnormal response procedures
    - Changes in emergency operating procedures (EOPs) and severe accident guidelines (SAGs)
    - Changes in licensed operator training program
    - Plant modifications

- TS amendments that change limiting conditions for operation and surveillance requirements
- Plant equipment failure rate and unavailability data in the maintenance rule database
- Plant preventive maintenance schedule (interval between maintenance or testing)
- Operating experience
- Plant events such as automatic or manual scrams and forced shutdown as reported in the licensee event reports
- Industry development in PSA technology, for example, treatments for HRA, common cause failures and uncertainties
- F&Os resulting from the peer review process

2. Review of PSA input should be documented. Items obtained from the review, which require a change to the PSA elements, should be tracked for incorporation into the PSA. Model errors or discrepancies identified in the PSA elements (mainly, system fault trees, event trees, and basic events database) should be documented in the corrective action program.

3. PSA Update/Upgrade:

   PSA updates/upgrades are based on significance of the changes made to the PSA elements instead of a fixed periodicity. PSA updates are performed periodically to ensure the PSA represents the as-built, as-operated plant. Updates of the PSA models reflect plant changes such as modifications, procedure changes, or plant performance (data). PSA upgrades represent the incorporation into the PSA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. This could include items such as new human error analysis methodology, new data update methods, new approaches to quantification or truncation, or new treatment of common cause failure.

   The extent of changes to PSA elements should be evaluated to determine if the changes result in a PSA upgrade. The BWROG guidelines are used for this evaluation.

   PSA upgrade receives a peer review, but is limited to the PSA elements that have just been upgraded.

4. Pending Changes - Delta CDF should be determined for each change made to the PSA elements. Significant changes should be incorporated into PSA update in a timely manner. Cumulative delta CDF estimates should be

determined for all changes made to the PSA elements pending incorporation into PSA update.

5. Use of Computer Codes - The computer codes used to support and to perform PSA analyses should be controlled in accordance with the software quality assurance program. A copy of computer codes, input files and output files should be stored in a secured site, to which the access should be limited to read-only.

6. Documentation Control - PSA documents should be maintained to reflect the current PSA model and the applications. A revision log should be completed to summarize the changes during PSA update. A checklist should be completed to document the verification of the revision of PSA documents.

7. Qualification - Revisions to PSA documents should be reviewed by a qualified reviewer. A qualified reviewer is defined as one who completed the performance and reliability qualification guide and meets the qualification for the PSA utility reviewer specified in the Nuclear Energy Institute (NEI) peer review guidance.

NRC Request:

e. ER Table E.3-2 presents the truncation limits used when quantifying the Internal Events and Fire PSA fault trees to be 1E-8 to 1E-14 with a footnote explaining that "the truncation limit was adjusted to assure sufficient capture of the contributing basic events." The meaning of and need for different truncation limits for fault trees, event trees and a category referred to as "Global" is not clear. Explain the basis for the truncation limits selected and the meaning of the entries shown in Table E.3-2. Clarify for which cases a truncation limit of 1E-8 versus a truncation limit of 1E-14 was used.

Energy Northwest Response to 1.e:

There are three distinct truncation limits that were used in the CGS PSA (including internal events, fire, and seismic models) computation, which can affect the final cutset equation. They are described below:

(1) Fault Tree Truncation

The unavailability of a safety function is typically represented as a functional equation, which is calculated from a fault tree (e.g., HPCS unavailability is represented as U1.EQN which is calculated from the fault tree HPCS.LGC. The unavailability of HPCS is calculated to be 2.1E-2). As part of the fault tree calculation, a truncation limit must be used, and it was carefully chosen

to maintain a balance between capturing all important cutsets, facilitating an efficient computation, and not exceeding software capabilities. One internal event Level 1 fault tree (U2.eqn) was set at a truncation limit of 2E-9 to limit the large number of cutsets. This is approximately eight orders of magnitude lower than the fault tree solution. All other internal event fault trees are solved at a 1E-10 truncation.

The range of truncation limits shown in Table E.3-2 reflects the results of exercising this described process. For example, in the internal events Level 2, the VIA2.EQN (an equation representing the combined core damage class VI-A(2) frequency) was calculated to be 2.34E-8 with a truncation limit of 1E-14, while another equation 3L1.EQN (representing "power not recovered prior to vessel failure") was calculated to be 5.88E-2 with a different truncation limit, 1E-8. By doing the above, the CGS PSA (including internal events, fire and seismic models) has generally maintained a four-order difference between functional equation value and its associated truncation limit. Exceptions are when simple fault tree functional equations exist where lesser differences are appropriate.

(2) Event Tree Truncation

The event tree truncation limit is for the merging of functional equations in the event tree. The factors that were considered in selecting an appropriate event tree truncation limit for each sequence were the same as those for the fault tree truncation limit described above. The CGS PSA has generally maintained a difference of five orders of magnitude between each individual sequence truncation and the final CDF for the internal events, fire, and seismic models. A few exceptions to the above occurred, such as in three of the turbine trip sequences and three of the manual shutdown sequences from internal events Level 1 due to software (WinNUPRA) maximum number of cutsets limitation. Also, four orders of magnitude truncation was utilized for fire Level 1 to maintain a practical solution time.

(3) Global Core Damage Truncation

After the individual sequences are all calculated, they are concatenated to yield a global core damage equation, which includes cutsets for all sequences from the associated initiators. In the CGS PSA, the global core damage equation truncation limit has been maintained the same as the event tree truncation limit to best preserve the cutset integrity.

NRC Request:

2) Provide the following information relative to the Level 2 analysis:

   a. ER Sections E.4.1.1 and E.4.3 state that containment event trees (CETs) were developed to model accident progression for accident sequences from the Level 1 internal events, fire, and seismic analyses. Provide a description of the CETs used in the SAMA analysis, discuss the basis for their construction, and clarify the relationship between the internal events, fire and seismic CETs.

Energy Northwest Response to 2.a:

The response is provided in three parts as follows:

(1) Provide a description of the CETs used in the SAMA analysis,
(2) Discuss the basis for the CET construction, and
(3) Clarify the relationship between the internal events, fire and seismic CETs

(1) Provide A Description of the CETs Used in the SAMA Analysis

Each Level 1 core damage sequence is assigned a PDS. The primary categorization used to bin the CGS Level 1 sequences is by accident type. That is, categories such as a) loss of containment heat removal (CHR), b) loss of coolant injection, and c) ATWS. Secondary binning consideration is done by initiator type (i.e., LOCA, LOOP, SBO), and tertiary binning is by the systems which may or may not be available to mitigate the accident after core uncovery. The fourth binning consideration is by the power and system recoverability.

A Level 2 CET is developed for each Level 1 PDS. Table E.4-1 provides a summary of the Level 1 PDSs where a Level 2 internal events CET is developed to support the containment performance evaluation. The quantified Level 2 CET end states are binned into four release categories for input to the SAMA analysis as shown in Table E.4-3 for internal events. Similar binning is performed for fires and seismic events.

(2) Discuss the basis for the CET construction

During construction of the CETs, two general rules were used as guidance because they assist in the processing and understanding of the information they contain:

- Events which exhibit the greatest amount of dependence with other events are shown as close to the beginning of the tree as possible, and
- Events are shown in the order in which they are generally expected to occur

Fault trees are used to evaluate each node for each accident sequence. The solution of the fault tree provides the relative likelihood with which each CET node branch represents the expected accident progression path. The fault tree nodes used in the construction of the CET accident sequences include the following:

- Containment isolated at time of core damage
- RPV depressurized prior to vessel failure
- Containment intact after vessel failure
- LPCI/LPCS recovered before containment failure
- Debris is cooled after vessel failure (ex-vessel)
- Shell failure due to high pressure melt ejection
- Residual heat removal (RHR) recovered
- Containment vent recovered
- Power conversion system (PCS) recovered for containment heat sink
- Containment failure mode is large
- Failure is in drywell

For LOOP scenarios, additional nodes are included for recovery of offsite power as follows:

- Power recovered prior to vessel failure
- Power recovered between vessel failure and containment failure

It is noted that not all fault tree nodes may be used for the construction of individual CETs. For example, if the containment is breached prior to core damage for a specific Level 1 PDS (e.g., ATWS scenario), then the node for containment isolation is not questioned in the associated Level 2 CET. Furthermore, if the Level 1 PDS involved loss of CHR, then recovery of RHR, containment vent, and PCS is not credited in the Level 2 CET.

The Level 2 CET accident sequence end states (release categories) are based on the following characteristics:

- Containment Failure Mode (Large or Small). Note that the CGS Level 2 CETs conservatively assume that all containment failures are Large in size.

- Time of Containment Failure (Early or Late). Early is defined as a release within 4 hours after a General Emergency is declared.

- Scrubbing of Fission Product (Scrubbed or Non-scrubbed). In general, scrubbing is assumed if the debris can be cooled or if the containment failure is in the wetwell.

The assignment of source terms to the release categories is supported by MAAP cases based on the inputs (i.e., containment failure location) and the results (magnitude, timing, and energy release to the environment).

(3) Clarify the relationship between the internal events, fire and seismic CETs

Separate fire and seismic CET structures are created based on the internal events CET structures. Table 2.a-1, below, provides a summary of the Level 2 CETs quantified for the Internal Events PSA, FPSA, and SPSA models.

Level 2 Fire CET Structures

The Level 2 fire CET structures and release categories are identical to the Level 2 internal events CET structures and release categories. Fire initiated events do not result in any unique containment challenges to the accident progression or the source terms to warrant any changes to the CET sequence structure or end state classification.

Refer to the sections below discussing "Release Magnitude Adjustments for SPSA" and "Release Timing Adjustments for SPSA" for additional details that also apply to the development of the Level 2 fire CETs.

Level 2 Seismic CET Structures

The Level 2 seismic CET structures and release categories are based on the Level 2 internal events CET structures and release categories. Adjustments to the Level 2 internal events CETs to create the Level 2 seismic CETs are discussed in the following sections.

Adjustments for Seismic-Induced Containment Structural Failure (CET for PDS 5)

The CGS SPSA directs accident sequences involving seismic-induced structural failure directly to the containment bypass accident class (Class 5), which directly results in LERF in the internal events Level 2 analysis. This is also assumed for seismic structural failures of the reactor building, containment, and reactor vessel pedestal. (Note that this assumption may be

conservative, depending on the type and extent of failure of the reactor building or reactor vessel pedestal. However, a more detailed analysis of these failures was not performed for this revision, and direct containment bypass was assumed.)

There are two other major structural failures included in seismic scenario for structural failure – the DG building and the radwaste control building. In neither of these cases would a direct bypass of containment occur. Therefore, these failures were separated from the direct bypass failures above and further analyzed. This resulted in developing a new CET for the Level 2 seismic Class 5 PDS.

Adjustments for AC Power Recovery

For the CGS SPSA, the offsite AC power recovery probabilities used in the Internal Events PSA are not applicable to the SPSA. In the 24-hour period after a seismic event, the probability of offsite AC recovery failure is very high, much higher than for internal events. As such, CET accident sequences that credit offsite AC power recovery need to be adjusted. For the purposes of the Level 2 calculation, the CGS SPSA assumes that offsite AC power recovery failure is 1.0 throughout the Level 2 accident sequences.

For the Level 2 internal events CETs where AC power recovery is credited, the sequences with successful AC power recovery have been eliminated for the Level 2 seismic CET structures.

Release Magnitude Adjustments for SPSA

The source term magnitude requires no adjustment for the SPSA. The release magnitude in the Internal Events PSA for a given accident is assessed based on the size of the containment failure and the presence of scrubbing. These characteristics apply equally whether the event is initiated by internal events or by a seismic event. Seismic scenarios that involve seismic-induced structural failures of the primary containment are directly classified as containment bypass scenarios. All other seismic scenarios that would involve accident progression phenomena are the same as those for internally initiated scenarios. Therefore, no changes are made to the internal events radionuclide release assessment with respect to the release magnitude for the Level 2 seismic CETs.

Release Timing Adjustments for Level 2 SPSA

Release sequences defined as "Late" for the Internal Events Level 2 PSA may be "Early" for seismic sequences given the seismic-induced impact on

evacuation activities (e.g., seismic-induced failure of evacuation infrastructure).

The following features were reviewed as part of the Level 2 SPSA development:

- Local roads and bridges
- Local population centers
- Emergency Plan

Based on the very low population within the 10-mile emergency planning zone (EPZ), the diverse road network, and the Emergency Plan and training/exercises, it is expected that evacuation within the 10-mile EPZ would be effective for a seismic event. Therefore, no adjustments for release timing are needed for the Level 2 SPSA assessment.

Summary of Level 2 Seismic CET Adjustments

Specific Level 2 seismic CET structures (e.g., Class 5 CET, CETs with AC power recovery) were modified when compared to the Level 2 internal events CETs. However, no adjustments to radionuclide release magnitude or release timing were required based on a detailed evaluation.

Table 2.a-1: Level 2 Containment Event Trees Used for Internal Events PSA, FPSA, and SPSA			
PDS	**Internal Events**	**Fire**	**Seismic**
1A1	Yes	(1)	N/A
1A2	Yes	Yes	N/A
1A3A	Yes	(1)	N/A
1A3B	Yes	Yes	Yes
1B0	Yes	Yes	Yes
1C	Yes	(1)	N/A
1G	Yes	Yes	(1)
1HA	Yes	(1)	N/A
1HB	Yes	Yes	Yes
2B	Yes	Yes	Yes
2D	Yes	Yes	Yes
3C	Yes	Yes	(1)
4BA	Yes	Yes	Yes
4BL	Yes	(1)	(1)
5	Yes	(1)	Yes
6A1A	Yes	(1)	N/A
6A1B	Yes	Yes	Yes
6A2	Yes	Yes	Yes
6B1	Yes	Yes	Yes
6B2A	Yes	(1)	N/A
6B2B	Yes	Yes	Yes

Note (1): Level 2 CET not quantified because Level 1 PDS is below truncation limit.

NRC Request:

b. ER Section E.4.1.1 states that CETs were developed for each plant damage state (PDS) and that quantification of the CETs was supported by fault tree analysis and assignment of split fractions. Clarify how probabilities were assigned to branches using split fractions for branches. In the response, specifically address how split fractions were developed for phenomenological branch points.

Energy Northwest Response to 2.b:

If a CET branch is based on a fault tree with individual failures, then the fault tree results are integrated with the appropriate CET branches and propagated through the CET sequence quantification. If a CET branch is based on point estimate probabilities (e.g., phenomenological branch points), then a single basic event is developed to model the split fraction for the branch point.

For example, the phenomenological branch points for the containment failure size (large or small) and location (drywell or wetwell) are point estimate probabilities based on a CGS plant specific structural analysis. The release size of the containment failure is treated as Large for all Level 2 release categories.

For other phenomenological branch points, fault tree models with individual failure modes were developed. For example, the CET branch point modeling "Containment Intact After Vessel Failure" is a fault tree developed to address the following failure modes:

- In vessel steam explosion
- Vessel blowdown
- Ex-vessel steam explosion
- Direct containment heating
- Hydrogen combustion

The probabilities used to model the CGS Level 2 phenomenological failure modes are consistent with established published PSA models (e.g., NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Appendix C.9, Vol. 2 and industry Level 2 PSA guidance documents (e.g., NUREG/CR-6595).

The point estimate branching fractions are conditioned by PDS inputs where appropriate. For example, point estimate branching fractions for offsite power recovery are conditioned by PDS input (i.e., the amount of credit for offsite power recovery in the Level 2 CET is conditional based on the amount of credit

previously assigned in the associated Level 1 PDS). It is noted that offsite power recovery is not credited in the fire or seismic Level 2 CET development.

Examples of point estimate branching fractions include the following:

- Containment isolated at time of core damage
- Power recovered prior to vessel failure (based on timing)
- Power recovered between vessel failure and containment failure (based on timing)
- Shell failure due to high pressure melt ejection
- Containment failure mode is large
- Failure is in drywell

Examples of fault tree modeled branch points include the following:

- Containment intact after vessel failure
- High pressure injection
- LPCI/LPCS recovered before containment failure
- Debris is cooled after vessel failure (ex-vessel)
- RHR recovered
- Containment vent recovered
- PCS recovered for containment heat sink
- Reactor vessel depressurized prior to containment failure

NRC Request:

c. ER Section E.5.5.1 lists peer review findings and other self-identified areas that are in progress for the next revision, and characterizes them as not expected to significantly alter the SAMA analysis findings. Yet, a number of the recommendations address non-conservatism in the Level 1 and 2 PSA model, including:

    - upgrading the LOCA outside containment modeling;
    - refining the inter-system (IS) LOCA modeling;
    - incorporating an initiating event for common cause failure (CCF) of both 125-VDC power divisions;
    - refining the impact of spray on equipment, the reactor core isolation cooling (RCIC) pump flood damage height, and flood isolation (human error probabilities) HEPs;
    - including certain early phenomena that can lead to large early relief frequency (LERF);
    - revising crew actions included in the LERF assessment;
    - accounting for potential environmental impacts in the survivability of

systems for Level 2 mitigation;
- reconsidering inclusion of source term scrubbing for non-LERF end states having no Modular Accident Analysis Program (MAAP) calculation.

Justify the conclusion that the unresolved findings are not expected to significantly alter the results of the SAMA analysis.

Energy Northwest Response to 2.c:

Section E.5.5.1 identifies areas of incompleteness that include conservatism and non-conservatism in the model. A treatment of both is required to properly evaluate any impact to the SAMA candidate results.

The above F&Os, which address areas of non-conservatism, were resolved in the upgraded internal events model. The results of the baseline comparison show a moderate increase in the internal events baseline CDF and a decrease in LERF.

The sensitivity study identified in response to RAI 1.a will assess the SAMA cases and provide any changes to the cost-benefit results that may result from the above Level 2 model incompleteness.

NRC Request:

d. Section E.6.5 and Table E.6-5 indicate the correlation between (MAAP) runs and release categories but does not provide the basis for the MAAP run selection. Provide information on the selection of the MAAP case for each release category, in particular how scenarios of less than dominant frequency but larger potential consequences were considered.

Energy Northwest Response to 2.d:

Section E.4.1.1 provides the following information:

"MAAP cases were binned into the appropriate Level 2 Release Category based on the inputs and results of the MAAP run (i.e., where containment failure was assumed and the resulting time and magnitude of the release).

For input into Level 3, representative MAAP cases were chosen primarily upon three criteria:

1. The MAAP case represents an accident class that would be expected to be included in the release category.

2. The MAAP case timing represents the appropriate timing characteristic of the release category (i.e., early vs. late).

3. The MAAP Cesium Iodide (CsI) release fraction is representative of the release category (i.e., > 0.1 for large [magnitude] release).

Where options exist for the potential assignment of various MAAP cases, cases were selected to include reasonable, but not undue conservatism. Thus, the LEN (large, early, non-scrubbed) case has a CsI release fraction of 0.2, which is well above the 0.1 threshold, but is less than a more extreme value of 0.5 as might be found for a break outside containment MAAP case."

Choosing a MAAP case to represent an entire release category can be difficult given the diverse number of scenarios that contribute to the overall release category frequency. The information in Section E.4.1.1 identifies the criteria used for selecting a representative MAAP case. Selecting a representative MAAP run was based on qualitative weighting factors such as the timing and magnitude of the initial release and the timing and magnitude of the total release. The methodology for choosing a representative MAAP case (e.g., performing an evaluation of the timing and magnitude of the radionuclide release) represents a best estimate evaluation of the Level 2 MAAP cases as input to the Level 3 offsite consequence evaluation.

For the LEN category, main steam line (MSL) failures with subsequent failure of the main steam isolation valves (MSIVs) to close (i.e., BOC) are not the dominant frequency contributor for internal events and fire. Loss of RPV Makeup and ATWS scenarios have a higher frequency contribution to the LEN category. Although the MSL failures scenarios would have a higher CsI release fraction than Loss of RPV Makeup or ATWS scenarios, based on a review of the CGS MAAP cases and the individual contributors to the LEN category, a large LOCA scenario with no injection available (MAAP run CGS08020) was chosen as a reasonable representation for the LEN release category. MAAP run CGS08020 has a CsI release fraction of 0.23 with the source term release beginning at approximately 3.9 hours due to drywell failure. The CGS Rev 6.2 Containment Performance Analysis defines "early" as a release that occurs in less than four (4) hours. MAAP run CGS08020 is a reasonable representation for the LEN release category because the CsI release fraction of 0.23 is above the "large" magnitude criterion of 0.1 and is of reasonably high magnitude to account for the contribution of low frequency scenarios with larger potential consequences (e.g., BOC scenarios). In addition, MAAP run CGS08020 satisfies the "early" release criterion for a source term release of less than four (4) hours.

Upgraded MAAP cases have been produced as part of the PSA upgrade recently completed. The sensitivity study identified in RAI 1.a will include specific MAAP

source terms for the MSL failures to assess if any SAMA cases would be impacted in their cost benefit evaluation.

The LLN (Large, Late, Non-scrubbed release) category is dominated by Loss of CHR and Loss of RPV Makeup scenarios. Although the Loss of CHR scenarios represent a higher contribution than the Loss of RPV Makeup scenarios, the Loss of CHR scenarios generally had a lower CsI release fraction than the Loss of RPV Makeup scenarios. Based on a review of the CGS MAAP cases and the individual contributors to the LLN category, an SBO scenario without RCIC injection and with late containment failure (MAAP run CGS08007) was chosen as a reasonable representation for the LLN release category. An SBO scenario without RCIC is similar to a Loss of RPV Makeup scenario in that core damage and vessel failure precede containment failure. An SBO is also similar to a Loss of CHR scenario because suppression pool cooling, containment venting, and other forms of CHR are unavailable. Based on the above characteristics, it is judged that an SBO scenario without RCIC has characteristics of both Loss of CHR and Loss of RPV Makeup scenarios. Therefore, using an SBO scenario without RCIC as the representative MAAP case for the LLN category, where the sequence frequency is dominated by Loss of CHR and Loss of RPV Makeup scenarios, is justified as appropriate. MAAP run CGS08007 has a CsI release fraction of 0.14 with the source term release beginning at approximately 7.1 hours due to drywell failure. MAAP run CGS08007 is a reasonable representation for the LLN release category because the CsI release fraction of 0.14 is above the "large" magnitude criterion of 0.1 and is of reasonably high magnitude to account for the contribution of low frequency scenarios with larger potential consequences. In addition, MAAP run CGS08007 satisfies the "late" release criterion for a source term release of greater than four (4) hours.

The LES (Large, Early, Scrubbed release) category does not have a quantifiable contribution to the CGS Rev 6.2 Level 2 radionuclide release frequency. Nevertheless, based on a review of the CGS MAAP cases, an unmitigated Turbine Trip ATWS scenario with early containment failure (MAAP run CGS08021) was chosen as a conservative representation for the LES release category. MAAP run CGS08021 has a CsI release fraction of 0.47 with the source term release beginning at approximately 3.4 hours due to failure in the wetwell airspace at 45 minutes and vessel failure at 3.4 hours. MAAP run CGS08021 is a conservative representation for the LES release category because the CsI release fraction of 0.47 is well above the "large" magnitude criterion of 0.1. In addition, MAAP run CGS08021 satisfies the "early" release criterion for a source term release of less than four (4) hours. Although the CsI release fraction is judged to be conservative for a scrubbed release, the LES category has a negligible contribution to the CGS Level 2 radionuclide release frequency and would not alter the conclusions of the SAMA evaluation.

The LLS (Large, Late, Scrubbed release) category is dominated by Loss of CHR and Loss of RPV Makeup scenarios. Although the Loss of CHR scenarios represent a higher contribution than the Loss of RPV Makeup scenarios, the Loss of CHR scenarios generally had a lower CsI release fraction than the Loss of RPV Makeup scenarios. Based on a review of the CGS MAAP cases, a Loss of RPV Makeup scenario with drywell sprays not used and late containment failure (MAAP run CGS08003B) was chosen as a reasonable representation for the LLS release category. MAAP run CGS08003B has a CsI release fraction of 0.10 with the source term release beginning at approximately 7.6 hours due to failure in the wetwell airspace. MAAP run CGS08003B is a conservative representation for the LLN release category because the CsI release fraction of 0.10 is equal to the "large" magnitude criterion of 0.1. In addition, MAAP run CGS08003B satisfies the "late" release criterion for a source term release of greater than four (4) hours. The CsI release fraction is judged to be conservative for a scrubbed release and, therefore, provides a small conservatism to the SAMA cost benefit analysis. (The LLS category contributes to approximately 10% of the CGS Rev 6.2 Level 2 radionuclide release frequency).

NRC Request:

e. Identify the version of MAAP used in the SAMA analysis.

Energy Northwest Response to 2.e:

The representative MAAP cases provided to model source terms as input to the MELCOR Accident Consequences Code System 2 (MACCS2) calculations are based on MAAP 4.0.4. These MAAP cases were specifically upgraded to the newer MAAP 4.0.4 for the SAMA analysis.

The MAAP calculations performed for the Level 1 and Level 2 PSA that were used to derive the Level 1 and Level 2 values for the SAMA cases were based on both MAAP 3.0B and MAAP 4.0.4. The use of the newer MAAP 4.0.4 for some MAAP cases was to meet other applications (Alternative Source Term, MSPI, etc).

NRC Request:

f. The ER does not provide an importance list of either Level 1 or Level 2 basic events and so it is not possible to ascertain the significance of recovery events or operator actions in the PSAs. Discuss the extent to which recovery of systems or operator actions following the onset of core damage is credited in the Level 2 assessment and how recovery is modeled.

Energy Northwest Response to 2.f:

Lists of system recoveries / operator actions credited in the Level 2 assessments for internal events, fire and seismic are provided below in Tables 2.f-1, 2.f-2 and 2.f-3 respectively. These important recoveries / actions were selected based on a threshold of RRW equal to 1.005 or greater.

For internal events, there are two important recoveries / operator actions: 1) failure to initiate automatic depressurization system (ADS) prior to vessel failure, and 2) failure to realign and initiate low pressure ECCS systems. A discussion of how the recoveries are modeled is provided in Table 2.f-1. The following SAMA cases are identified:

1. SAMA candidate CC-11 proposes to allow operators to inhibit ADS for non-ATWS scenarios, to reduce the potential for unintended depressurization. This enhancement has been implemented at CGS. When in the EOPs, operators inhibit the automatic operation of ADS using the ADS DIV 1(2) INHIBIT switches to preclude injection of large amounts of cold unborated water, which may result in core damage. Manual actions are then taken to depressurize the reactor when required. As a result of this enhancement, the potential for not initiating ADS prior to vessel failure increased in importance. A review of operator training determined that significant emphasis is made on the importance of initiating ADS if previously inhibited in accordance with the EOPs. The response to RAI 5.e identifies a general operator procedural and training-related SAMA case to be evaluated in the sensitivity study.

2. Failure to realign and initiate low pressure ECCS systems is a SAG action when reactor water level can be restored and maintained above top of active fuel. Operator training and clear, detailed procedures have been implemented at CGS. The relatively high importance of this high stress action is appropriately modeled. SAMA candidate CC-01 assesses the installation of an independent active or passive high pressure injection system, which provides a mitigation alternative for this operator action. This candidate bounds the risk considerations for recovery of low pressure ECCS systems. CC-01 was found to be non-cost-effective. A sensitivity study identified in the

response to RAI 5.e for enhancement to operator procedures and training will assess improvement of this action.

Important recoveries / operator actions for fire Level 2 include the two that are important for internal events, as well as the failure to recover offsite power three hours prior to vessel failure. Fire damage that causes LOOP is postulated to not be recoverable for the FPSA. SAMA candidate AC/DC-14 assesses the installation of a buried offsite power source. If the routing of this offsite source was sufficiently diverse and routed within the plant with sufficient separation, this alternate offsite source could be used to recover from a fire caused loss of the credited offsite power sources. AC/DC-14 was found to be non-cost-effective. Alternately, SAMA Candidate AC/DC-12 (improve 4.16 kV bus crosstie ability) to crosstie DG-3 to a Division 1 or Division 2 emergency bus during a LOOP has already been implemented at CGS.

For seismic, important recoveries / operator actions consist of offsite power recovery actions. Offsite power recovery following a seismic event is postulated to not be credible for the SPSA Level 2. AC/DC-14 assesses the installation of a buried offsite power source. If the power source was located sufficiently distant and not nearer than the plant to known geological fault structures, this mitigation alternative could provide the ability to recover an alternate offsite power source. The difference in transmission (underground transmission versus overhead towers) would provide added seismic ruggedness. AC/DC-14 was found to be non-cost-effective. Alternately, SAMA candidate AC/DC-12 (improve 4.16 kV bus crosstie ability) to crosstie DG-3 to a Division 1 or Division 2 emergency bus during a LOOP has already been implemented at CGS.

Table 2.f-1: Level 2 Internal Events Important System Recoveries / Operator Actions					
EVENT NAME	**DESCRIPTION**	**POINT EST.**	**F-V IMPORT**	**RRW**	**DISCUSSION**
ADSHUMNSTARTH3LL	Operator fails to respond in time to initiate ADS prior to vessel failure (VF)	1.00E-01	1.36E-01	1.158	SAG-1 (CF/P-1) "RPV Pressure"; concurrent procedure is PPM 5.1.1 "RPV Control".
LPSHUMNRESTORE	Failure to realign and start low pressure systems	1.00E-01	5.73E-02	1.061	SAG-1-TAB-D "Injection into the RPV can be Restored and Maintained"; concurrent procedure is PPM 5.1.1 "RPV Control".

Table 2.f-2: Level 2 Fire Important System Recoveries / Operator Actions					
EVENT NAME	**DESCRIPTION**	**POINT EST.**	**F-V IMPORT**	**RRW**	**DISCUSSION**
ADSHUMNSTARTH3LL	Operator fails to respond in time to initiate ADS prior to VF	1.00E-01	2.29E-01	1.297	SAG-1 (CF/P-1) "RPV Pressure"; concurrent procedure is PPM 5.1.1 "RPV Control".
LPSHUMNRESTORE	Failure to realign and start low pressure systems	1.00E-01	1.26E-01	1.145	SAG-1-TAB-D "Injection into the RPV can be Restored and Maintained"; concurrent procedure is PPM 5.1.1 "RPV Control".
NPWRVF-6A1	No power recovery prior to VF (~3 hr)	1.00E+00	1.02E-01	1.113	Fire damage that causes LOOP is postulated to not be recoverable for the FPSA.

Table 2.f-3: Level 2 Seismic Important System Recoveries / Operator Actions					
EVENT NAME		**POINT EST.**	**F-V IMPORT**	**RRW**	**DISCUSSION**
NPWRCF-1A3	No power recovery prior to containment failure (CF) (~15 hr) for PDS 1A3	1.00E+00	1.53E-01	1.181	Offsite power recovery following a seismic event postulated to not be possible for seismic Level 1 and Level 2.
NPWRVF-1A3	No power recovery prior to VF (~3 hr) for PDS 1A3	1.00E+00	1.53E-01	1.181	Offsite power recovery following a seismic event postulated to not be possible for seismic Level 1 and Level 2.
NPWRVF-6A1	No power recovery prior to VF (~3 hr) for PDS 6A1	1.00E+00	6.39E-03	1.006	Offsite power recovery following a seismic event postulated to not be possible for seismic Level 1 and Level 2.
NPWRCF-6A1	No power recovery prior to CF (~19 hr) for PDS 6A1	1.00E+00	6.14E-03	1.006	Offsite power recovery following a seismic event postulated to not be possible for seismic Level 1 and Level 2.

NRC Request:

3) Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

   a. ER Section E.3.2.2 states that the seismic hazard analysis used for the seismic PSA is the same as submitted for the CGS Individual Plant Examination of External Events (IPEEE) except for an extrapolation from the maximum peak ground acceleration to 1.5g. The seismic hazard analysis used for the IPEEE was developed in 1994 and documented in "*Probabilistic Seismic Hazard Analysis WNP-2 Nuclear Power Plant Hanford Washington*". Justify the use of the seismic PSA model given: (1) since then the U.S. Geological Survey (USGS) has updated its assessment of seismic hazards across the U.S. including Washington State, (2) seismic hazard analysis was performed specifically for the Hanford area in 1994 which is documented in WHC-SD-W236A-TI-016, *Seismic Design Spectra 200 West and East Areas DOE Hanford Site, Washington*", to provide better evaluation of subsurface materials and (3) work was performed in 2005 which is documented in PNNL-15089, "*Site-Specific Seismic Response Model for the Waste Treatment Plant, Hanford Washington*" that better characterizes the effect from deep layers of sediments "interbedded" with basalt. Address whether consideration of the more current seismic hazard analysis could impact the results of the SAMA analysis (both SAMA identification and SAMA evaluation).

Energy Northwest Response to 3.a:

The 1994 seismic hazard analysis used at CGS was developed by Geomatrix Consultants for Energy Northwest. A similar hazard model was used by Geomatrix to evaluate the United States Department of Energy (USDOE) facilities located elsewhere on the Hanford site in 1994 (Seismic Design Spectra 200 West and East Areas DOE Hanford Site, Washington, WHC-SD-W236A-TI-016, referenced in the RAI). This USDOE work was superseded by a revised report in 1996 (Probabilistic Seismic Hazard Analysis DOE Hanford Site, Washington, Report Number WHC-SD-W236A-TI-002, Rev. 1, dated February 1996). The application of this hazard model to each different Hanford site requires revision of the distances between the site being evaluated and the known and postulated seismic sources contained in the model. Site specific hazard curves are developed for each site evaluated.

The CGS site is located approximately 10 miles southeast of the USDOE Waste Treatment Plant (WTP) that is located adjacent to the 200 East area of the Hanford site. The CGS site has distinct differences from the WTP due to its increased distance from nearby seismic sources and different foundation conditions. The more northerly WTP site is located in close proximity to Central

fault on Gable Mountain (about 6 km north of the WTP site) and is incrementally closer to the other Yakima fold seismic sources compared to the more distant CGS site location to the southeast. Other factors being equal, increased distance from a seismic source tends to reduce the expected ground motions at a site.

At the CGS site the soil structure is thicker than at the WTP. However, the deeper basalt flows and alternating sedimentary interbed sequence is similar between the two sites. The combined thickness of the Hanford and Ringold soil formations is approximately 380 feet thick at the WTP (PNNL-16652, Figure 2.2) in contrast to approximately 480 feet at CGS (FSAR Figure 2.5-28). In general the upper Hanford formation is thinner (250' WTP vs. 65' CGS), and the Ringold sediment section is thicker at the CGS site (130' WTP vs. 415' CGS).

A large body of geotechnical data was gathered by Energy Northwest during the initial site investigations for CGS and the adjacent WNP-1 and WNP-4 plant sites. These investigations included the acquisition of extensive velocity data for the combined sites. During initial plant licensing for CGS (FSAR Appendix 2.5Q), Energy Northwest performed comparative site response studies using the soil velocity profile for the CGS site and typical firm alluvial soil profiles representative of California strong motion recording sites. For frequencies above about 3 Hz, the California sites used in the site-specific spectrum showed more amplification than the CGS site (FSAR Appendix 2.5Q, Figures 361.17-23 and 24). The conclusion of that analysis was that the empirical strong motion data from firm alluvial sites in California was appropriate for use at the CGS site (FSAR Appendix 2.5Q and the NRC Safety Evaluation Report, NUREG-0892, Supplement 1). This conclusion was adopted for the CGS 1994 seismic hazard study.

During a design review, the Defense Nuclear Facilities Safety Board questioned the original WTP seismic design (based on their 1996 hazard analysis) regarding the assumptions used in developing the original seismic criteria and the adequacy of the WTP site geotechnical surveys. To allow the project to proceed until new data could be acquired, a very conservative interim seismic design spectrum was developed that was documented in PNNL-15089 (2005, referenced in the RAI). Geotechnical work was initiated in 2005 to obtain new WTP site-specific data. This work was primarily directed at obtaining new shear wave velocity data including an improved understanding of the velocity contrast between the basalt flows and intervening sedimentary interbeds.

In 2007 USDOE issued another round of reports based on the new data provided by the site-specific geotechnical investigations (see reports PNNL-16407, PNNL-16652, and PNNL-16653). In general, the overall ground motion response was less than the interim values estimated in 2005 due to the new velocity

information that indicated a greater shear wave velocity contrast between the basalts and interbeds and new data from the sediments that reflected greater damping.

Of importance to the CGS site is their conclusion regarding the 1996 ground motion models which was based on the seismic hazard model adopted from CGS. They concluded in PNNL-16653 (Updated Site Response Analysis for the Waste Treatment Plant, DOE Hanford Site, Washington, 2007, page 37), that "the hazard results obtained using the new ground motion models at the WTP site are similar to those obtained using the 1996 set of ground motion models." The relative amplification function (ratio of Hanford / California response) for the WTP site based on the updated site response model is generally below 1.0 (i.e., WTP site response is less than predicted using California recordings) with the exception of minor isolated peaks at 2, 4 and 20 Hz (see PNNL-16653, Figure 33). This is a large reduction over the interim relative amplification factors developed for the WTP in 2005 (PNNL-15089, Figure 3.3.9) where the Hanford response is predicted to be greater than the California data for most frequencies greater than about 1 Hz.

The United States Geological Survey (USGS) recently updated (2008) its assessment of seismic hazard for the United States. The results of this national program provide an opportunity for an updated independent validation of the results determined by Geomatrix for the CGS site. The USGS website offers its results either in the form of a contour map or more directly by the gridded data set that was used to construct the maps. The grid file (0.05 degree increment) was used to avoid interpolation of the small scale map contours. The USGS hazards results from two of the grid files (for 119.35° W, 46.50° N) are compared with the mean results from the Geomatrix 1994 report for the CGS site in Table 3.a-1 below. The Geomatrix (CGS) values are similar but slightly larger than those calculated by the USGS.

Table 3.a-1: Comparison of USGS and Geomatrix (CGS) Data

Study	PGA for T = 500 years (10% in 50 years)	PGA for T = 2500 years (2% in 50 years)
USGS 2008	0.072 g	0.169 g
Geomatrix 1994	0.081 g	0.178 g

Although differences exist in the methods used to develop the individual site response models for different Hanford facilities, Energy Northwest concludes that the recent site-specific work performed by USDOE for the WTP validates earlier conclusions regarding the applicability of the California strong motion database to the estimation of ground motions at Hanford. Further, it should be noted that the other aspects of the hazard analysis such as fault locations, earthquake magnitudes and frequencies and attenuation relationships were not reexamined

in the course of the USDOE studies (PNNL-15089, Summary statement, page iv) and thus those fundamental components of the earlier hazard studies have not changed and would still apply. Comparison of the mean CGS hazard to the independently determined 2008 USGS hazard calculations verifies that the CGS model is conservatively predicting an appropriate ground motion for the CGS site. Accordingly, Energy Northwest concludes that the 1994 seismic hazard study still provides an adequate seismic input to the PSA models to effectively identify all relevant SAMA candidates.

NRC Request:

b. ER Section E.3.2.1 states that the guidance in NUREG/CR-6850 was used to update the IPEEE Fire PSA to the current CGS Fire PSA. This section also states that extinguishment and propagation split fractions and likelihood information from NSAC/178L was used. Clarify to what extent NUREG/CR-6850 is used to update the IPEEE Fire PSA and, in particular, whether use of information from nuclear safety analysis center (NSAC)/178L represents a deviation from NUREG/CR-6850 or was used to augment NUREG/CR-6850. It is also stated that, in general, the Fire PSA results dominate the SAMA risk evaluation due to conservatisms from NUREG/CR-6850. Describe these conservatisms and how they compare to those that were in the IPEEE. Discuss whether there was any attempt to reduce these "conservatisms" by plant-specific analysis, e.g., fire modeling of specific configurations. If not, discuss whether any potential non-fire insights could be "masked" by this conservatism.

Energy Northwest Response to 3.b:

Use of NUREG/CR-6850 was limited to only the refinement of electrical hot short probabilities. Information from NSAC/178L "Fire Event Database for U.S. Nuclear Power Plants" was used to develop extinguishment and propagation split fractions. This approach is judged to be sound, but does not follow the approaches suggested by NUREG/CR-6850.

The conservatisms in the IPEEE were addressed in Rev 6.2 and previous revisions by a significant effort devoted to refinement of the dominant fire scenarios, particularly in the area of more refined assessments of cable selection (to remedy errors and conservatisms in identifying which cables impact FPSA components) and cable routing (through the examination of the raceway drawings to locate specific raceway locations). Plant-specific fire modeling was performed and used; however, this effort could be expanded. Additional conservatisms can be removed from the FPSA model through the refinement of existing fire scenarios and development of additional detailed fire scenarios.

In terms of level of detail, the Rev. 6.2 FPSA is judged to be good, relative to other PSAs of its vintage and used in previous LRA approved applications. The PSA doesn't meet NUREG/CR-6850 guidance in areas such as HRA and fire growth modeling, but it has incorporated significant improvements since the IPEEE.

However, until the FPSA is upgraded, fire-related risk insights may be masked. The FPSA risk profile could change once the upgrades / refinements / updates have been performed, but only conjecture can be made at this time.

There was no attempt to reduce conservatisms in the Revision 6.2 PSA model when performing the SAMA analysis by the use of additional or selective refinement of the FPSA elements.

Non-fire risk insights are judged to not be masked for the SAMA evaluation. To develop the SAMA cases, risk insights from each of the PSAs (internal events, fire and seismic) were examined individually as well as in the aggregate, and SAMA cases were developed accordingly. Table 3.b-1 provides examples of the sources for these insights.

Table 3.b-1: Sources of Risk Insights for SAMA Candidates

SAMA Candidate	Risk Insight Source
CB-01, AT-13, AT-14	Internal Events PSA
FR-03, FR-07a and FR-07b	FPSA
SR-01 and SR-03	SPSA
AC/DC-27	Internal Events PSA and FPSA
AC/DC-01, AT-05, AT-07	Internal Events PSA and SPSA
AC/DC-10, AC/DC-23, AC/DC-28, AC/DC-29, CC-01, CC-03b, CC-20, CP-01, CW-02, CW-03, CW-07, HV-02	Internal Events PSA, FPSA and SPSA

NRC Request:

c. ER Section E.3.2.1 explains that for the screening fire event trees, scenarios in which equipment and/or cables were lost due to a fire (i.e., "loss scenarios") were simplified into loss of worst-case or all equipment and cables in a fire compartment. The section also explains later that hot shorts that could spuriously actuate components to undesired configurations were considered for the unscreened sequences. Explain how potentially screening out sequences that might have contained risk significant hot short events affects the results of the fire PSA and in turn the SAMA evaluation.

Energy Northwest Response to 3.c:

No sequences were screened out; the screening was to determine those sequences that required further development before quantification. Screening fire event trees for each compartment were developed incorporating extinguishment and propagation split fractions from NSAC/178L, Revision 1, automatic suppression when applicable, and likelihood of plant trip. If the fire is suppressed or self extinguishes early, the frequency for this condition transfers to a conditional event tree in which the single worst failure for that compartment is applied. If not, the frequency for this condition transfers to a conditional event tree in which a full compartment burnout is modeled. Therefore, each compartment initially has a fire initiating event tree, and two conditional fire event trees for single equipment or cable or compartment losses. The conditional fire event trees are either turbine trip or loss of feedwater event trees, and quantified as appropriate for the compartment losses, including all applicable hot short events.

These screening fire event trees are conservative, and are used for initial evaluation to identify those compartments that require more detailed modeling. They were quantified as an initial step in the FPSA plant response model development.

After quantification of the screening event trees, those compartments found to have an initial CDF greater than 5.0E-7/yr were analyzed in more detail to be more realistic. Typically, the approach was to identify more scenarios for each compartment and model each scenario with its own conditional fire event tree.

The quantified results for those compartments with initial CDF below 5.0E-7/yr per compartment were not refined further, and the cutsets and CDF contributions for these are retained and reported in the fire CDF.

NRC Request:

d. ER Section E.5.5.2 states that a number of recommended improvements to the fire PSA from the 2004 fire PSA peer review were outstanding but that none were expected to significantly alter the SAMA analysis findings. Yet, a number of the recommendations appear to be non-conservative in the fire PSA such as:

- The PSA does not include all the cable routing for all conduits installed in the plant;
- A hot short probability of 0.3 is used, which implicitly assumes all circuit failures are intracable for multi-conductor cables protected by controlled power transformers (from NUREG/C-6850);

- A transformer fire scenario must be re-evaluated for switchgear Division 2 to remove non-conservatism from current modeling;
- The current fire PSA credits systems or trains that fire-related plant procedures instruct operators to defeat. In addition, a human error probability of 0.1 is used to indicate the need to restart the Division A or HPCS equipment;
- The approach to quantify the approximately 130 individual hot short events corresponding to single spurious actuations from the internal events PSA is cited as capturing most, but not all, of the multiple spurious operations (MSOs) that need to be modeled.

In light of these apparent non-conservatisms, justify the conclusion that the unresolved findings are not expected to significantly alter the results of the SAMA analysis.

Energy Northwest Response to 3.d:

Rev. 6.2 of the FPSA has resolved all of the significant F&Os from the 2004 peer review, with the exception of one that will reduce model conservatism and thus not impact the SAMA evaluation (see response to RAI 1.c.ii). The items noted in RAI 3.d were not generated from the 2004 peer review but were from a technical evaluation of the Rev. 6.2 FPSA after completion of the model (Reference RAI 1.c.iii Technical Review 7). These were items identified for future incorporation into the FPSA and may indicate areas of model incompleteness.

It is our judgment that these items won't significantly impact the SAMA analysis findings. It is believed that the re-analysis of the SAMA candidates associated with RAI 6.j using the 95% percentile will apply an increased multiplication factor that should be sufficient to account for these areas of modeling incompleteness, including potential cumulative impacts. This factor increases the baseline FPSA CDF to 2.3E-5/rx-year, which is judged to be an adequate upper bound to fire risk. The specific considerations are:

(i) The electronic database used to select and locate cables does not include all conduit locations. The FPSA cable selection effort does appropriately treat cable terminal locations and thus fire damage to cables installed in conduits that travel from cable trays to terminal equipment within the same physical analysis unit is captured. However, for conduits not captured by terminal locations, significant additional walkdown and analysis would be required to completely address this modeling incompleteness. The model incompleteness is judged to be encompassed by the response to RAI 6.j.

(ii) For FPSA Revision 6.2, the hot short probability of 0.1 was increased to 0.3 to address an F&O associated with the 2004 fire peer review. This treatment

did not take into account the specific circuit and cabling configurations. Based on a review of NUREG/CR-6850 guidance, the 0.3 probability was judged to be an appropriate representative value, and this assumption and failure probability is judged to be reasonable for the LRA. The potential modeling uncertainty is accounted for by the response to RAI 6.j.

(iii) A transformer fire scenario must be re-evaluated for Division 2. This is an area of incompleteness in the FPSA. In Rev 6.2, the fire scenario selection for the Division 1 switchgear room was updated to remove the apparent non-conservatism. However, these refinements actually reduced the total CDF contribution for Division 1 switchgear room fires from 2.0E-6/rx-yr to 1.0E-6/rx-yr. Model changes for Division 2 were thus deferred. We have carried the description for Division 2 transformer modeling issues as non-conservative until the upgrade of the FPSA is performed. Based on the outcome of the Division 1 refinements, enhancements of the Division 2 FPSA modeling, including inclusion of the transformer fire, are anticipated to not significantly alter the results of the SAMA analysis.

(iv) The current FPSA credits systems or trains that fire-related plant procedures instruct operators to defeat. This issue is related to modeling the degree of discretion that operators have in use of equipment and systems that might be impacted by a fire in the area of that equipment or system. This issue has been recently re-evaluated as part of the non-conformance review to RG 1.189 Rev. 2 "Fire Protection for Nuclear Power Plants" and has been resolved. The operators have discretion to continue using a system in service during the fire until the fire causes safe shutdown parameter degradation or the visible damage to vital plant equipment or cabling. The modeling in the Rev. 6.2 FPSA is compatible with this development.

(v) The hot short events modeled in the FPSA correspond to all single spurious actuations modeled by the Internal Events PSA and include, for example, spurious closure of a valve in the RCIC flow path, spurious closure of a valve in the HPCS flow path, or spurious closure of a valve in the RHR flow path to the suppression pool. The FPSA sequence quantification captures all combinations of these 130 individual hot short events that contribute to the accident sequences above the quantitative truncation limit. This approach captures most, but not all, of the MSOs that may need to be modeled in the PSA. The model incompleteness is judged to be accounted for by the response to RAI 6.j.

Areas of non-conservatism have been identified for the Rev. 6.2 FPSA. In addition, areas of conservatism are also known to exist, particularly in the area of further refinements in fire scenario selection. The future upgrade of the FPSA will address these, and the eventual net risk impact of these refinements is

unknown. However, at this time it is believed that the re-analysis of the SAMA candidates associated with RAI 6.j using the 95th percentile will apply an increased multiplication factor that should be sufficient to account for these areas of modeling incompleteness.

Based on the above, these areas of model incompleteness, when adjusted per the RAI 6.j request, are not expected to significantly alter the results of the SAMA analysis.

NRC Request:

e. ER Table E.3-7 shows the top 20 risk dominant "fire compartments" ordered by their contribution to the total CDF based on the most current version of the fire PSA. The IPEEE also presents a similar list of "fire compartments" ordered by percent contribution to the total CDF. Explain why the total CDF due to fire is different (7.40E-6/yr versus 9.16E-6/yr) and why the risk dominant fire compartments and the ordering of their risk contribution are different.

Energy Northwest Response to 3.e:

The decrease in total CDF and re-ordering of dominant fire compartments in Rev. 6.2 is a result of significant refinements in 1) cable selection that remedied errors and reduced conservatisms in identifying which cables impact FPSA components, and 2) cable routing that revised specific raceway locations through the examination of the raceway drawings. The Rev. 6.2 FPSA includes more detailed fire scenarios and updated fire frequencies.

NRC Request:

f. ER Section E.8.5 states that "the benefit from the "other" hazard group contribution was conservatively estimated to be equivalent to that of internal events." Clarify the basis for this assumption.

Energy Northwest Response to 3.f:

The bases for the assumption are as follows:

- Some of these "other" external events are captured quantitatively in the LOOP contributor assessment for weather and other related LOOP contributors.

- An assessment was made in the IPEEE of the contribution to CDF from other events. The IPEEE used the screening approach per Generic Letter (GL)

88-20 "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)" and found the other events all contributed less than 1E-6/rx-year to the baseline CDF.

- The assumption that the contribution to total CDF from other events is equal to the CDF for internal events, which is over four times the CDF estimated in the IPEEE, is sufficiently conservative to not understate the CDF contribution benefit for evaluated SAMA candidates due to other external events.

- NEI 05-01 "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document" provides guidance on applying an external events multiplier on the maximum SAMA benefit derived from fire and seismic risk. This approach was not necessary for fire and seismic, but the use of a multiplier of unity for other external events is conceptually consistent with the NEI guidance.

NRC Request:

g. ER Table E.5-1 reports a PSA Revision 3 Level 2 release frequency (9.94E-06/yr) that is significantly higher than the previous revisions while the release frequency for Revision 1 is significantly less than for Revision 0. Discuss the major model changes driving these differences between versions. Also, explain the major reasons for the CDF decrease by roughly a factor of 2.5 from Rev. 4.2 to Rev. 5.0.

Energy Northwest Response to 3.g:

The changes that have occurred over the past 20 years of PSA model development at CGS have been influenced by a number of factors, chief among them are the maturity of the methodology through NRC and industry consensus standards and technical reports. These have allowed the industry to refine the internal events modeling and bring the plant's PSA to a more realistic level.

The detailed history of these changes to the model is difficult to detail without significant research of archived documentation. However, the following information is known that may provide useful insights:

- The original submittal in response to GL 88-20 was withdrawn due to model incompleteness and conservatism primarily in the Level 1 model. The Revision 0 response to GL 88-20 was produced primarily by internal staff that had minimal experience in PSA development. The Revision 0 model was withdrawn based on independent review by experts. The decreases in CDF and the increase in Level 2 were the outcome of primarily more realistic modeling through the input of outside consultants.

- The Level 2 value of 1.07E-6 listed in Table E.5-1 for Revision 1 is incorrect. The July 1994 Revision 1 Report of the IPE reported the correct value of 1.07E-5. Major model changes that affected the Revision 3 (September 1997) Level 2 results were the changes in the PDS contribution from the Level 1 model changes listed in Table E.5-1. This change in CDF PDSs directly affected the Level 2 results. The Level 1 and the Level 2 baseline models decreased by approximately the same amount.

- Revision 5.0 was a major revision that focused on the DG Completion Time TS submittal. This revision was performed with significant consultant support and to the trial use criteria of RG 1.200. Many enhancements were made as reported in Table E.5-1. Generally, the reduction in CDF was through removal of conservatism that existed in the model. The primary model features were related to revising the LOOP event tree sequences, applying industry power recovery probabilities, and performing realistic battery calculations that extended the life of the batteries and associated RCIC operating time. The extended RCIC operating time allowed lower non-recovery of offsite power probabilities. There were additional conservatisms in the model that were addressed, including adding additional battery chargers, revising ECCS pump room HVAC dependencies, and adding reactor building HVAC (normal HVAC) to the model. Of significance was that this PSA revision was part of the pilot program for RG 1.200 and both an external peer review team and an NRC inspection provided F&Os that were resolved in the DG Completion Time submittal through incorporation of the issue in the model or by other acceptable means such as sensitivity analyses. These F&Os identified conservative and non-conservative factors, as is discussed in the Response to RAI 1.c.

NRC Request:

h. ER Section E.5.5.3 does not identify any reviews of the Seismic PSA. Identify any internal and external reviews of the Seismic PSA and provide an assessment of the impact of any unresolved findings on the SAMA evaluation.

Energy Northwest Response to 3.h:

Revision 0 of the SPSA was focused on development of a seismic model using ANSI/ANS 58.21-2003 "External Events in PRA Methodology". Revision 0 was not issued for any application. A self-assessment was performed on the Revision 0 SPSA against ANSI/ANS 58.21-2003 and identified four Supporting Requirements that were not met (excludes documentation-only findings). In addition, the assessment noted that a peer review was not performed. These issues were seen as not significant for incorporation into Revision 1 and a

decision was made to wait for a peer review to incorporate the issues. When work was completed on the Revision 6.2 Internal Events PSA model, the SPSA model development was subsequently updated.

Table 3.h-1 provides the self assessment information and evaluates the impact on the SAMA evaluation. Revision 1 was completed in February 2007 and was available for the SAMA analysis in 2008. Revision 0 and Revision 1 were performed by internal and external PSA engineers with expertise in seismic modeling. No review has been performed to assess the CGS SPSA against ANSI/ANS 58.21-2007, ASME/ANS RA-Sa–2009 Rev. 1, or RG 1.200 Rev. 2.

The industry is currently piloting the seismic standard and future enhancements are anticipated in the methodology. The use of the CGS SPSA model provides a realistic ability to assess seismic risk for the SAMA evaluation. The use of the plant specific model in lieu of the NEI 05-01 multipliers is judged to be a superior method for assessing SAMA benefits.

Table 3.h-1: Summary of CGS SPSA Capability Per ANS Standard ANSI/ANS-58.21-2003		
Supporting Technical Requirement	Self Assessment Issue	SAMA Impact Assessment
HA-D2	Expert elicitation was not used to characterize the ground motion for this study.	The ground motion was characterized by Geomatrix. Recent studies have confirmed the CGS site seismic characterization to be adequate. See RAI 3.a.
SA-C1	Given the current state of the ANS External Event Standard (i.e., little differentiation among capability category thresholds), it is unclear whether the standard will in the future require greater justifications and/or sensitivities (i.e., beyond the existing statements in the SPSA) regarding the assumptions related to assumed failures (e.g., condensate storage tank [CST]). In any event, a sensitivity case that addresses the fragility of the CST (rather than simply assuming failure probability of 1.0) should be quantified and included in the SPSA documentation.	The revised SAMA cases using the multiplication factor derived from 95% percentile in RAI 6.j is judged sufficient to assure that sensitivity cases addressing the fragility of certain features (e.g., CST) are adequately considered and no impact to the SAMA analysis cost benefit results would occur.
SA-E6	The ANS External Event PRA Standard is clear on the importance of performing various sensitivity studies to address a variety of modeling and parametric uncertainties, such as: assumed HEP impacts; fragility correlation assumptions; etc. Although the CGS SPSA includes some quantitative sensitivity studies, additional cases are assumed required to meet the intent of this SR. Refer to Table 2-4 of EPRI TR-1003121 for example sensitivity cases.	The revised SAMA cases using the multiplication factor derived from 95% percentile in RAI 6.j is judged sufficient to assure that sensitivity cases addressing the fragility of certain features (e.g., CST) are adequately considered and no impact to the SAMA analysis cost benefit results would occur.

Table 3.h-1: Summary of CGS SPSA Capability Per ANS Standard ANSI/ANS-58.21-2003		
Supporting Technical Requirement	Self Assessment Issue	SAMA Impact Assessment
FR-C6	Soil-structure interaction (SSI) analysis was performed by Geomatrix for the WNP-2 IPEEE; however, it is unclear whether the detailed aspects (e.g., minimum value of C_V no less than 0.5, mean and standard deviation of low strain shear modulus for every soil layer, etc.) of this SR are included in that analysis. This issue needs to be verified to confirm the CGS SPSA capability for this technical requirement.	Documentation of the low strain shear modulus for every soil layer is not considered significant. The ground motion was characterized by Geomatrix. Recent studies have confirmed the CGS site seismic characterization to be adequate. See RAI 3.a.
HA-J1 FR-G1	(2) The 2004 (updated in 2006) CGS SPSA does not include a peer review, when one is completed it should be documented.	Peer review would be to the RG 1.200 Rev 2 and combined ASME/ANS RA-Sa-2009 after pilot plant completion and standard update. Impact to SAMA analysis is indeterminate.

NRC Request:

i. ER Section E.4.2 states that the Level 2 release categories for the Fire PSA indicate ~88% to be late and, therefore, non-contributors to LERF. Describe the phenomenology that causes more of the releases for fire occurring later than for internal events. Also, comparing Tables E.4-1 and E.4-4 indicates that the main difference between the Internal Events and Fire LERFs arises from PDS 1A3B, 1C, 4BA, 4BL, 5, 6A1A, and 6A1B. 1C is a flood PDS, so that difference is apparent. Others involve loss of offsite power (LOOP), ATWS, ISLOCA, and SBO, for which all but the LOOP LERFs are lower for Fire than Internal Events. Since fires often induce the same sequences as internal events, but with greater probability for fire-induced vs. random failures, describe the phenomenology that causes the LERF for most of these fire PDSs to be lower than for internal events, especially since Fire CDF is greater than Internal Events CDF.

Energy Northwest Response to 3.i:

Per Table E.4-3, approximately 47% of the Level 2 Internal Events PSA release categories are late while approximately 89% of the Level 2 FPSA release categories are late. This difference is because the Level 1 FPSA has a significantly higher contribution to long term Loss of DHR scenarios (non-LERF contributors) than the Level 1 Internal Events PSA.

Table E.4-1 shows that the Level 1 Internal Events PSA contribution to Loss of DHR scenarios (PDSs 1B0, 2B, and 2D) is approximately 17%. In comparison, Table E.4-4 shows that the Level 1 FPSA contribution to Loss of DHR scenarios (PDSs 1B0, 2B, 2C, and 2D) is approximately 53%. The significantly higher Level 1 FPSA contribution to long term Loss of DHR scenarios, when compared to the Level 1 Internal Events PSA, results in a higher contribution to Level 2 FPSA late release contributors.

Table 3.i-1 provides a comparison of the Level 1 PDS frequencies for the Internal Events PSA and the FPSA based on the information in Tables E.4-1 and E.4-4.

The CGS Level 1 FPSA model has a higher contribution to Loss of DHR scenarios due to any of the following reasons, or a combination of them:

- Fire initiating events may fail or impact use of the main condenser for heat removal
- Fire initiating events may fail or impact use of containment venting for heat removal
- Fire initiating events may fail a single division of suppression pool cooling

Table 3.i-2 provides a comparison of the Level 2 LERF frequencies by PDS for the Internal Events PSA and the FPSA based on the information in Tables E.4-1 and E.4-4. The RAI identifies differences in the internal events and fire LERF results for PDSs 1A3B, 1C, 4BA, 4BL, 5, 6A1A, and 6A1B. The primary contributors to the differences in the LERF results for the identified PDSs are as follows:

- PDS 1A3B – The FPSA LERF value is **higher** because the Level 2 FPSA does not credit recovery of HPCS.

- PDS 1C – The FPSA LERF value is **lower** because the FPSA does not model fire-induced flooding scenarios that contribute to this PDS.

- PDS 4BA – The FPSA LERF value is **lower** because the FPSA does not model fire-induced ATWS events.

- PDS 4BL – The FPSA LERF value is **lower** because the FPSA does not model fire-induced ATWS events.

- PDS 5 – The FPSA LERF value is **lower** because the FPSA does not include fire-induced containment bypass events.

- PDS 6A1A – The FPSA LERF value is **lower** because the FPSA does not credit recovery of HPCS(1).

- PDS 6A1B – The FPSA LERF value is **lower** because the Level 1 FPSA PDS is lower. The conditional LERF split fraction is 6.8E-2 for both the Internal Events PSA and the FPSA.

Note (1): PDS 1A3A and 6A1A represent scenarios where HPCS is recoverable after core damage. Given that HPCS recovery is not credited in the FPSA model, PDS 1A3A and 6A1A are not applicable. Therefore, PDS 1A3A and 6A1A have a frequency of 0.0/yr in Table E.4-4.

Table 3.i-1: Comparison of Level 1 PDSs for CGS Internal Events PSA and FPSA				
PDS	**Level 1 Internal Events PSA**		**Level 1 FPSA**	
	Frequency (/yr)	**Percent**	**Frequency (/yr)**	**Percent**
1A1	5.10E-08	1.1%	0.00E+00	0.0%
1A2	8.40E-07	17.5%	8.30E-07	11.2%
1A3A	4.70E-08	1.0%	0.00E+00	0.0%
1A3B	9.80E-08	2.0%	3.20E-07	4.3%
1B0(1)	3.10E-07	6.5%	2.40E-06	32.4%
1C	1.50E-07	3.1%	0.00E+00	0.0%
1G	4.90E-07	10.2%	1.60E-06	21.6%
1HA	3.50E-08	0.7%	0.00E+00	0.0%
1HB	4.70E-08	1.0%	7.70E-08	1.0%
2B(1)	1.60E-09	0.0%	2.80E-08	0.4%
2C(1)	N/A	N/A	1.50E-06	20.3%
2D(1)	5.10E-07	10.6%	0.00E+00	0.0%
3C	3.00E-07	6.3%	N/A	N/A
4BA	1.10E-07	2.3%	2.70E-10	0.0%
4BL	6.40E-08	1.3%	0.00E+00	0.0%
5	1.50E-07	3.1%	0.00E+00	0.0%
6A1A	3.00E-07	6.3%	0.00E+00	0.0%
6A1B	7.40E-07	15.4%	3.70E-07	5.0%
6A2	2.30E-08	0.5%	7.60E-08	1.0%
6B1	3.30E-07	6.9%	2.70E-07	3.6%
6B2A	5.70E-08	1.2%	0.00E+00	0.0%
6B2B	1.40E-07	2.9%	3.70E-08	0.5%
Total	4.80E-06	100.0%	7.40E-06	100.0%

Note (1): PDSs associated with long term Loss of DHR scenarios.

Table 3.i-2: Comparison of Level 2 LERF by PDS for CGS Internal Events PSA and FPSA				
PDS	**Level 2 LERF Internal Events PSA**		**Level 2 LERF FPSA**	
	Frequency (/yr)	**Percent**	**Frequency (/yr)**	**Percent**
1A1	5.50E-09	0.8%	0.00E+00	0.0%
1A2	9.10E-08	13.9%	9.00E-08	36.6%
1A3A	2.90E-10	0.0%	0.00E+00	0.0%
1A3B	**4.30E-09**	**0.7%**	**1.30E-07**	**52.8%**
1B0(1)	0.00E+00	0.0%	0.00E+00	0.0%
1C	**1.50E-07**	**23.0%**	**0.00E+00**	**0.0%**
1G	3.80E-10	0.1%	1.23E-09	0.5%
1HA	2.70E-11	0.0%	0.00E+00	0.0%
1HB	3.60E-11	0.0%	5.97E-11	0.0%
2B(1)	0.00E+00	0.0%	0.00E+00	0.0%
2C(1)	N/A	N/A	0.00E+00	0.0%
2D(1)	0.00E+00	0.0%	0.00E+00	0.0%
3C	2.30E-10	0.0%	N/A	N/A
4BA	**1.10E-07**	**16.8%**	**2.70E-10**	**0.1%**
4BL	**6.40E-08**	**9.8%**	**0.00E+00**	**0.0%**
5	**1.50E-07**	**23.0%**	**0.00E+00**	**0.0%**
6A1A	**2.00E-08**	**3.1%**	**0.00E+00**	**0.0%**
6A1B	**5.00E-08**	**7.7%**	**2.50E-08**	**10.2%**
6A2	0.00E+00	0.0%	0.00E+00	0.0%
6B1	0.00E+00	0.0%	0.00E+00	0.0%
6B2A	0.00E+00	0.0%	0.00E+00	0.0%
6B2B	0.00E+00	0.0%	0.00E+00	0.0%
Total	6.53E-07	100.0%	2.46E-07	100.0%

Note (1): PDSs associated with long term Loss of DHR scenarios.

NRC Request:

4) Provide the following information concerning the Level 3 analysis:

   a) ER Tables E.6-2 and E.6-3 do not appear to be consistent. For example, the year 2045 population in Table E.6-3 is 655,617, a more than 70% increase compared to the year 2030 population in Table E.6-2 (383,828) and considerably greater than the stated growth rate of 14.2%/decade. Table E.6-2 indicates a leveling off of the population by 2030 as well. Clarify this apparent discrepancy and provide the reference year used for escalating population to year 2045. Also, the CGS Final Safety Analysis Report (FSAR) Table 2.1-1 population growth rate is closer to 4.5%/decade. Given that the FSAR is stated to be the source of the population data used in the SAMA analysis, explain the differences in population growth rate between the ER and the FSAR.

Energy Northwest Response to 4.a:

Tables E.6-2 and E.6-3 were included for different purposes. The information in Table E.6-2 was future population estimates as extracted from the FSAR (Table 2.1-1). These data were included to show the decreasing trend in population growth rate within a 50-mile radius of CGS. However, the population data taken from Table 2.1-2 of the FSAR (adjusted for transient population) was escalated based on Washington State population and presented in Table E.6-1. Therefore, the escalated population values do not match the predicted values in Table E.6-2. The decreasing growth rate trend in Table E.6-2 is meant to underscore the conservative assumption to use a constant escalation rate (per decade) to estimate the 2045 population (in Table E.6-3).

NRC Request:

   b) The population distribution within sectors in ER Table E.6-3 is not consistent with the FSAR Table 2.1-1 distribution. The N sector at 40-50 miles shows ~100% growth from the FSAR year 2030 compared to year 2045, while other sectors show ~50%. Clarify the apparent different growth rates within sectors given the stated growth rate of 14.2%/decade.

Energy Northwest Response to 4.b:

The 2030 population data in the FSAR Table 2.1-1 is an estimate of the projected population growth in a 50-mile radius around CGS. The methodology for the population estimate in the FSAR is different from the conservative population escalation approach used to develop the 2045 population used as input to the MACCS2 code (Table E.6-3). The growth rate of 14.2% per decade was used to

estimate the 2045 population using the 2000 population (census data from FSAR Table 2.1-2) as the starting point. Comparing the percent increase of the population from 2000 (with one escalation scheme) and the projected population in 2030 (with a different escalation scheme), particularly on a per-sector basis, is likely to result in inconsistent results (as noted in the RAI). As stated in the response to RAI 4.a, the inclusion of Table E.6-2 (data taken from FSAR Table 2.1-1) showing the 2030 data is to show the decreasing trend in population growth rate to underscore the conservative assumption of using a constant escalation rate (per decade) to estimate the 2045 population (in Table E.6-3).

NRC Request:

c) ER Section E.6.3 states that the year 2006 meteorological data used in the base case contained the least amount of unusable data. Describe how gaps or missing data were filled and how unusable data were replaced with usable data.

Energy Northwest Response to 4.c:

Unusable meteorological data were designated with a value of "-99" for wind speed, wind direction, delta temperature, or precipitation. No actual unusable meteorological data were provided. The approach used to "fill in" the gaps for all the meteorological data was:

- For spans of unusable data less than 10 hours, the valid data on either side of the span were averaged. This average number was entered for all data points of the unusable span.

- For spans of unusable data greater than or equal to 10 hours, data from the previous and subsequent hours was duplicated and used for the unusable data spans. For example, a 24-hour span of unusable data used the previous 12 hours of valid data and the subsequent 12 hours of valid data.

  The approach was reviewed and found to be acceptable by a meteorologist, who also reviewed the "filled in" meteorological data.

  Example (< 10 hours)

Julian Day	Hour	Wind Speed
31	11	58
31	12	**-99**
31	13	53

For 31/12, an average of the before and after values ((58+53)/2 = 55) was used.

Julian Day	Hour	Wind Speed
31	11	58
31	12	**55**
31	13	53

Example (≥ 10 hours)

Data from 249/14 through 250/11 were identified as "-99" (unusable). The values from **249/14 through 249/24** (first half) are replaced by data from 249/3 through 249/13; the values from **250/1 through 250/11** (second half) are replaced by data from 250/12 through 250/22.

Julian Day	Hour	Wind Speed
249	3	7
249	4	5
249	5	7
249	6	12
249	7	15
249	8	6
249	9	6
249	10	9
249	11	14
249	12	6
249	13	15
249	14	**7**
249	15	**5**
249	16	**7**
249	17	**12**
249	18	**15**
249	19	**6**
249	20	**6**
249	21	**9**
249	22	**14**
249	23	**6**
249	24	**15**
250	1	**22**
250	2	**10**
250	3	**21**

250	4	**22**
250	5	**23**
250	6	**27**
250	7	**29**
250	8	**36**
250	9	**33**
250	10	**34**
250	11	**34**
250	12	22
250	13	10
250	14	21
250	15	22
250	16	23
250	17	27
250	18	29
250	19	36
250	20	33
250	21	34
250	22	34

NRC Request:

d) ER Section E.6.4 states that the MELCOR Accident Consequence Code System 2 (MAACS2) default growing season was assumed. Clarify how this relates to the length of the growing season for the CGS locality and provide an assessment of the impacts of this assumption on the SAMA evaluation.

Energy Northwest Response to 4.d:

The growing season used by the SAMA evaluation is the default growing season specified by MACCS2. Information on at least one food product grown in the eastern Washington region within the 50-mile radius of CGS confirms that the regional growing season is longer than the assumed default growing season. To provide an assessment of the impact to the SAMA evaluation of a longer growing season, a MACCS2 sensitivity case with a conservatively defined growing season of February 1 to November 30 (302 days) for all seven crop groups was performed.

The results of the sensitivity case showed no change in the consequence metrics from the base case. The results for CGS are not sensitive to the growing season length. Therefore, the assumed MACCS2 default growing season lengths for the seven crop groups are deemed acceptable for the MACCS2 calculation to support CGS's SAMA evaluation.

NRC Request:

e) ER Sections E.6.5 and E.6.6.3, and Table E.6-6 identify the OALARM time to declaration of a General Emergency and the assumed time delay for evacuation. Section E.6.6.3 identifies that "it is not clear" what the 50 minute delay represents. One would typically expect some fixed delay time in declaration of an emergency requiring evacuation, and then an evacuation time based on various parameters (time of day, season, weather, etc). Provide additional discussion of the evacuation analysis and interpretation of the time delay.

Energy Northwest Response to 4.e:

Two different parameters are discussed in Section E.6.5 and Section E.6.6.3. In Section E.6.5, there is a discussion about the MACCS2 parameter OALARM, which is identified as "warning time." This is often interpreted as the time to declare a General Emergency. This warning time, OALARM, is different and separate from the time delay for evacuation discussed in Section E.6.6.3. In fact, an evacuation could not occur until t=OALARM. For the MACCS2 model, OALARM is estimated by the time-to-core-uncovery (on a per release category basis), as determined in the MAAP run (for a specific release category).

Section E.6.6.2 discusses the evacuation speed. The development of evacuation speed considered normal and adverse weather conditions. Sensitivity cases were performed on the evacuation speed; these are discussed in Section E.7.2.4.

Section E.6.6.3 discusses the evacuation delay time, which is the time it takes to begin evacuation after a declaration of General Emergency. This estimate is based on information from an Energy Northwest report "Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone, Columbia Generating Station" dated April 2005. The information in this report provides evacuation delay times as a function of population types, and in some cases, as a function of the time of day (daytime versus nighttime). Upon additional review, a footnote on Table 5-1 of this report indicated that the delay times were based on a 15-minute notification time. Nonetheless, to account in the uncertainty of the notification time, a sensitivity case was run to evaluate the impact of an additional 60 minutes (estimated average delay on mobilization of the offsite emergency response agency upon notification of an emergency) on the results. As the consequence metrics to support the SAMA analysis were unchanged with the increased delay time, the analysis was conducted with the information as obtained from the report.

NRC Request:

f) ER Section 6.8 describes that much of the economic data used for the MACCS2 analyses were site-specific. However, the MACCS2 economic data provided in Table E.6-10 appear to be taken directly from MACCS2. Clarify whether a cost escalation or inflation factor was applied to MACCS2 default values to bring the costs up to year 2008 dollars.

Energy Northwest Response to 4.f:

The MACCS2 economic data provided in Table E.6-10 are taken from the MACCS2 sample case (default values) and used for input to the CHRONC Module. These economic values have not been subject to either a cost escalation or inflation factor.

No escalation was applied because the results of the CHRONC module (economic impact) are used only on a relative basis (delta-cost) in the SAMA analysis. When the total benefit is determined, the economic impact for the SAMA candidate under evaluation is subtracted from the economic impact of the maximum benefit case to ascertain the "increment of benefit." If the economic impact for the maximum benefit case increases, then so will the economic impact for the SAMA candidate, leaving the delta-cost relatively unaffected. Also, the large contributions from onsite costs and replacement power likely mitigate any variations of offsite costs affected by the parameters in Table E.6-10.

Nonetheless, a sensitivity case was performed to confirm the above assertions. The sensitivity case escalated the MACCS2 parameters POPCST, CDFRM0, CDNFRM and DLBCST (from Table E.6-10) by 4.1% (an estimate of the inflation rate from 1993 to 2008, as discussed in the response to RAI 6.i). The cost-benefit analysis shows that there is only about a 0.5% increase in the total benefit for each of the SAMA candidate cases when the new cost vector was used. Therefore, the absolute values of these parameters were not deemed to be crucial and were not escalated.

NRC Request:

g) Sensitivity case S1 uses the year 2060 population extrapolation to represent a high population growth case for the year 2045. The stated growth rate of 14.2%/decade would yield ~22% increase from year 2045 to 2060 (from 665,617 to ~812,000). This is then a sensitivity case that increases the year 2045 population by 22%. Clarify whether this interpretation is correct. If this estimated population is not correct, explain how the calculation is performed. It also is stated that the state-wide growth is conservatively applied to the

county growths. However, the two most populous nearby counties (Benton and Franklin) had growth rates greater than the state average. Clarify.

Energy Northwest Response to 4.g:

1. The interpretation that sensitivity case S1 effectively increases the population by 22% is correct.

2. The growth rates for Benton and Franklin Counties can be estimated from the 2000 to 2006 census website data. The decade growth rate for Benton is slightly greater than the state rate, while the growth rate for Franklin is significantly greater. However, the population of Franklin County is only 10% of the surrounding county's population. The 14.2%/decade growth rate is applied to all of the counties surrounding CGS. Further, the county population data available is a narrow snapshot (in time) of the population trend in Washington State. For that reason, the Washington State population trend was presented in Table E.6-2 showing a steady predicted decreasing trend (reducing to a growth rate of only 1.03%/decade) in 2030. On that basis, predicting the population to 2045 was considered conservative (over-estimating the population) by using a constant growth rate of 14.2%/decade.

NRC Request:

h) ER Section E.7.2.3 Case A1 identifies that the "rain rate boundary condition was set at 0.0 mm/hour for the base case" is a bounding conservative value. Previous SAMA analyses have shown that assuming perpetual rainfall in the last segments (40-50 mile radius) is the most conservative assumption when estimating population dose and cost risk. Clarify the statement that no rainfall in the last boundary segments is a conservative assumption. In addition, provide the assumptions used for each of the meteorological boundary parameters used in this sensitivity case.

Energy Northwest Response to 4.h:

The maximum rainfall in an hour from the 2006 meteorological data was 0.14 inches. The MACCS2 parameter BNDRN for the boundary rain rate is in millimeters/hours (accordingly, 0.14 inches/hours = 3.6 millimeters/hours). A sensitivity case was run with this parameter to test the sensitivity of the assumption that zero rainfall is a conservative value.

The original rationale was that with no rainfall, any radioactivity particulates would not get washed out and would be available to cause continued radiation dose to the population and environment. The sensitivity case showed no consequence metrics were affected by the increase in rainfall as the boundary

condition. This neither proves nor disproves the assumption of conservatism. However, with METCOD=2 (use meteorological sampling option of "weather bin sampling"), and the boundary conditions only being used if 120 hours of recorded weather data does not transport the last plume through the limiting spatial interval, it is likely the boundary meteorological parameters do not play a significant role in the results.

For BNDMXH (boundary value for the mixing layer height), the value 1140 meters was used, which is the average of the eight values (per seasonal and time of day).

For IBDSTB (boundary weather stability class index), a stability class of F (value=6) was used based on a statistical analysis of the meteorological data showing that it is the most likely stability class.

For BNDWND (boundary wind speed), the value 2.0 meters/second was used based on a statistical analysis of the meteorological data showing that it is the most likely wind speed.

NRC Request:

5) Provide the following with regard to the SAMA identification and screening process:

   a. ER Section E.9.1 identifies 12 industry SAMA analyses that were reviewed and Table E.9-3 identifies which of these analyses was the source for many of the CGS SAMA candidates. However, it is unclear why some of the cost-beneficial SAMAs from the industry analyses do not appear to be included in Table E.9-3, i.e., providing redundant ventilation for residual heat removal pump rooms, high pressure core spray pump room, and reactor core isolation cooling pump room (Nine Mile Point SAMAs U2-23a-c), reduce unit cooler contribution to emergency diesel generator unavailability (Nine Mile Point SAMAs U2-221a-b), etc. Provide an assessment of the applicability of each of the cost-beneficial SAMAs from the 12 industry SAMA analyses to CGS and a further evaluation for those that are applicable.

Energy Northwest Response to 5.a:

A review was undertaken of all the cost-beneficial SAMA candidates provided in the industry SAMA analyses referenced. In the 12 industry SAMA analyses listed in Table E.9-3, 72 SAMA candidates were identified. All were reviewed for applicability to CGS.

Twenty-one (21) candidates were not applicable to CGS because the candidate was site specific or based on a different BWR design (i.e., HPCI versus HPCS, ice condenser verses suppression pool, etc). Twenty (20) candidates have already been implemented at CGS. Ten (10) candidates have already been considered in Table E.9-3 and the disposition in Table E.10-1 was for further evaluation and the results provided.

The remaining industry cost-beneficial candidates, along with the 4 candidates specifically identified in the RAI, are listed in Table 5.a-1, and an assessment of the applicability to CGS is provided. This evaluation identified additional SAMA candidates to be added to the sensitivity study to be performed in response to RAI 1.a.

Table 5.a-1: Review of Industry BWR Cost Effective SAMA Candidates		
Potential SAMA Candidate	**Response**	**Screening Criteria**
Develop procedures to control feedwater flow without DC power	CGS loss of DC power from DC Bus B1-7 will result in both the turbine-driven reactor feedwater (RFW) pumps tripping and a low low reactor water level will result in trip of the MSIVs. With MSIVs closed, injection would be via RCIC or HPCS. Feedwater would likely not be an immediate option due to reopening of the MSIVs. Late recovery may be feasible. A sensitivity study will be performed to verify the risk improvement of this option.	Criterion C Considered for Further Evaluation To be evaluated in the PSA 7.1 sensitivity study
Revision of the operating procedure to provide additional space cooling via the use of portable equipment or blocking doors for RHR pump rooms	This was already implemented in procedures at CGS	Criterion B Already Implemented at CGS
Revision of the operating procedure to provide additional space cooling via the use of portable equipment or blocking doors for HPCS pump room	This was already implemented in procedures at CGS	Criterion B Already Implemented at CGS
Revision of the operating procedure to provide additional space cooling via the use of portable equipment or blocking doors for the RCIC pump room	This was already implemented in procedures at CGS	Criterion B Already Implemented at CGS
Hard pipe diesel fire pump to vessel	CGS procedures provide the capability to connect fire protection water to the suction side of condensate booster pump for supply to the vessel. This is through	Criterion C Considered for Further Evaluation

Table 5.a-1: Review of Industry BWR Cost Effective SAMA Candidates		
Potential SAMA Candidate	**Response**	**Screening Criteria**
	staged fire hose and dedicated connections. Converting this to a hard pipe installation will be further evaluated.	To be evaluated in the PSA 7.1 sensitivity study
Increase testing frequency of DG unit cooler	SAMA candidate increases testing of unit coolers for the DG control panel rooms. The coolers are cleaned and inspected every 2 years. Additionally, as part of the heat exchanger and cooling coil performance monitoring program, other service water (SW) coolers are periodically checked as leading equipment for fouling of the heat exchangers. The two spray pond pump house HVAC coiling cools are the monitored cooling coils for the room coolers supplied by the SW system. They are tested annually for thermal performance. Should thermal performance degradation be detected, other SW heat exchangers would be inspected through the corrective action program. Additionally, the SW supplied heat exchangers and cooling coils are part of the SW chemical control program. Monthly surveillance tests of the DG causes the SW cooling coils to be used. The heat transfer capability of the CGS DG cooling coils would be checked if any degradation in cooling was noticed. Further, as provided in the Aging Management Program, testing of a sample of the cooling coils for the diesel, radwaste, spray pond and reactor building cooling units will be performed prior to entering the period of extended operation. Very little additional benefit would occur by adding the DG cooling coils to the periodic performance testing as part of this existing	Criterion E Very Low Benefit

Table 5.a-1: Review of Industry BWR Cost Effective SAMA Candidates		
Potential SAMA Candidate	**Response**	**Screening Criteria**
	reliability program.	
Improve control building flooding scenarios	CGS recently identified control building flooding as a potential high risk contributor during the PSA upgrade and established actions within the corrective action program. A potential SAMA for improvements in detection and mitigation for control building flooding will be investigated further.	Criterion C Considered for Further Evaluation To be evaluated in the PSA 7.1 sensitivity study
Proceduralize <u>all</u> potential 4 kV crossties	CGS can crosstie DG-3 to either emergency bus SM-7 or SM-8 by procedure. Using DG-3 hardware to crosstie Division 1 and Division 2 is possible but overload potential of the DG would reduce the risk benefit value. This pathway is not of high benefit when the DG-3 cross-connect is available. Prior NRC approval would be required. Backfeeding the HPCS system with SM-8 would provide a third power source for HPCS. A SAMA candidate to evaluate this is proposed for use in the EOP/SAGs.	Criterion B Already Implemented at CGS for Division 3 to Division 1 or Division 2 See AC/DC 12 Criterion C Considered for Further Evaluation To be based on PSA 7.1 sensitivity study for crosstie from Division 2 to Division 3
Diverse swing DG air compressor	Each emergency DG is provided with separate, independent starting air systems. Each starting air system for DG-1 or 2 has two electric air compressors and sufficient air receivers to provide for a total of five DG starts from each air header. For additional flexibility in the system, the air receivers can be interconnected so that either of the two (2) air circuits is capable of	Criterion B Already Implemented at CGS For DG-3 Criterion E Very Low Benefit for DG-1 and DG-2

Table 5.a-1: Review of Industry BWR Cost Effective SAMA Candidates		
Potential SAMA Candidate	**Response**	**Screening Criteria**
	starting the diesel engine in the event of a major component failure in one circuit. The DG-3 starting air system has two air compressors one electric and one diesel, and two air receivers; each of which has sufficient capacity for three engine starts. This SAMA candidate is already implemented at CGS for DG-3. The diverse (diesel swing compressor) for DG-1 and DG-2 would provide minimal increase in risk improvement.	
Enhance alternate injection reliability by including RHR SW and fire water crosstie in maintenance program	This will be included in the CGS list of SAMA candidates to be considered for further evaluation.	Criterion C Considered for Further Evaluation To be based on PSA 7.1 sensitivity study
Add emergency level control sensor and control valve to the hotwell	This SAMA candidate would add redundant hotwell level controls to the condenser for additional assurance that hotwell level inventory would remain viable for condensate and feedwater injection. CGS design includes dual hotwell level controllers with back up controllers and an alternate hotwell level control. Additionally, hotwell level control can be performed manually through direct control of makeup valves from the control room. This SAMA would add additional sensors and control valves powered from the emergency power. However, if the BOP power source is not available, the condensate and condensate booster pumps would not be available. Thus, this SAMA candidate would only provide benefit for failed	Criterion E Very Low Benefit

Table 5.a-1: Review of Industry BWR Cost Effective SAMA Candidates		
Potential SAMA Candidate	**Response**	**Screening Criteria**
	level controllers and failure to control level manually. With the current level of redundancy and the ability to control hotwell level manually, very low benefit is achievable.	
Modify plant procedures to open doors in emergency DG building on receipt of high temperature alarm	Plant procedures allow for this action. However, opening doors and installing fans have limited effectiveness in the DG room areas. The engine/generator area is able to operate up to 130°F. However, the electrical equipment panel room immediately adjacent to the engine area is limited to 122°F. Ventilation flow paths are limited in the electrical equipment area to avoid drawing in higher engine room temperature.	Criterion E Very Low Benefit
Increase fire pump house building integrity to withstand higher winds so the fire system will be capable of withstanding a severe weather event	One diesel fire pump is located in the circulation water building and the other is in its own pump house. This SAMA candidate will be evaluated in the PSA 7.1 sensitivity study.	Criterion C Considered for Further Evaluation To be based on PSA 7.1 sensitivity study
Protect transformers from explosive failure	The CGS startup transformer and backup transformer are not in close proximity. However, there are the step-up main transformers and auxiliary transformers that separate them. Although CGS transformers are protected with sudden pressure relays to mitigate rapid pressure increases from resulting in an explosion, should they fail to respond fast enough there is a possibility that missiles or fire generated from a	Criterion C Considered for Further Evaluation To be based on the PSA 7.1 sensitivity study

Table 5.a-1: Review of Industry BWR Cost Effective SAMA Candidates		
Potential SAMA Candidate	**Response**	**Screening Criteria**
	transformer explosion could impact other transformers and potentially their incoming associated power lines. Although the cost for explosion protection is expected to be significantly greater than the maximum benefit, a SAMA case will be considered.	
Relocate relief valve cables, circuitry, and components, as well as other modifications to ensure one train of core spray remains unaffected by fire.	RAI 5.l requested more information in protecting one train of RHR and SW. This industry SAMA is similar. A SAMA cost-benefit analysis is being performed in response to RAI 5.l.	See response to RAI 5.l
Increase operator training on systems and operator actions determined to be important from the PSA	A SAMA sensitivity study discussed in response to RAI 5.e will be performed to enhance operator awareness and training for time critical high risk operator actions.	Criterion C Considered for Further Evaluation To be evaluated in the PSA 7.1 sensitivity study

NRC Request:

b. The conclusion to ER Section 9.1 is that "no additional candidates were identified by the review of the supplements to NUREG-1437." However, Table E.9-3 identifies NUREG-1437 Supplements 21 and 30 as the source for SAMAs AT-13 and AT-14, respectively. Clarify this discrepancy.

Energy Northwest Response to 5.b:

SAMA candidates AT-13 and AT-14 were added based on the review of the supplements to NUREG-1437. The wording in Section E.9.1 was modified in Amendment 10 to identify the source of these candidates.

NRC Request:

c. Table E.9-3 identifies four SAMA candidates based on CGS PSA insights (all others were identified from review of industry data). Section E.3.1 discusses Level 1 basic event importance analysis and presents high level insights but does not provide the results of a basic event importance analysis that show the potential risk reduction associated with specific basic events. A level 2 importance analysis is not discussed or presented. Provide a basic events importance list, in decreasing order of risk reduction worth (RRW), for the Level 1 and Level 2 internal, fire and seismic PSA results that includes a description of each basic event, identifies the RRW and probability of each basic event, and identifies the SAMA(s) that address each basic event and how. Provide the information for all basic events having an RRW benefit value greater than the minimum cost of a procedure change at CGS.

Energy Northwest Response to 5.c:

Table E.9-3 was created by first assembling a generic list of industry SAMA candidates. This was an extensive list of potential SAMA candidates based on BWR experience; therefore, many of these SAMA candidates had application to CGS. These SAMA candidates were evaluated based on CGS cutset and importance results as presented in Tables E.9-1 and E.9-2, with RRW values presented as a basis in Table E.9-2 for systems or components that were modeled in the PSA. It is proposed that the requested list of basic event importance values with descriptions and any additionally identified related SAMA candidates will be based on the results of the sensitivity study identified in the response to RAI 1.a.

NRC Request:

d. ER Table E.3-8 presents the results of an RRW importance analysis of the top 30 basic events from the Fire PSA. No SAMAs were identified to address any of the basic events in this table. Similarly, Table E.3-7 identifies the contribution to fire CDF from each of the CGS fire compartments, but no SAMAs were identified to address the most risk important compartments. Identify and evaluate SAMAs to address all of the basic events and fire compartments having an RRW benefit value greater than the minimum cost of a procedure change at CGS. In the response, describe how the SAMA addresses the basic event, fire compartment, or equipment in the fire compartment.

Energy Northwest Response to 5.d:

The requested benefit based on "An *RRW benefit value greater than the minimum cost of a procedure change at CGS"* is variable depending on the significance of the basic event and the degree of improvement that the procedure change obtains. The potential for a procedure change alone to improve a basic events importance for detection or mitigation of fire hazards is remote but can serve as a screening value. The basic cost of a procedure change alone is estimated to be $12,000. This estimate assumes no engineering, licensing action, etc. is required. Assuming maximum benefit is achieved, the RRW associated with fire is 1.015.

Table 5.d-1 identifies the important fire compartments (RRW greater than or equal to 1.015) and basic events and examines existing and potential SAMA candidates that address them.

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
1	EFF	Early extinguishment fails (fixed ignition source)	2.649	This early extinguishment probability is empirical based on a review of the fire events database. New potential SAMA candidates identified below to install early fire detection systems in significant fire compartments will address this basic event.
2	CF-FAILS-INJECT	Injection fails due to containment failure	1.289	This injection loss following containment failure probability is based on structural analysis. SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
3	ETF	Early extinguishment fails (transient ignition source)	1.282	This early extinguishment probability is empirical based on a review of the fire events database. New SAMA candidates identified below to install early fire detection systems in significant fire compartments will address this basic event.
4	FR1J	Fire in PAU R1J - Reactor building 522 elevation	1.192	A potential SAMA candidate is to install early detection in this compartment.
5	FW14	Fire in division 1 switchgear room	1.155	A potential SAMA candidate is to install early detection in this compartment.
6	FW04	Fire in division 1 electrical equipment room	1.128	A potential SAMA candidate is to install early detection in this compartment.
7	HS-CIAV-MO30A	CIA-V-30A failure due to hot short	1.127	This event affects the ability to open the safety relief valves (SRVs). The FPSA conservatively does not credit the air accumulators installed at each of the SRVs. Therefore, this basic event is judged to not be a realistic contribution to risk. No SAMA candidates identified.

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
8	RHR----B----T3LL	RHR train B unavailable due to test or maintenance	1.125	SAMA case CP-01 "Install an independent method of suppression pool cooling" addresses this basic event.
9	HS-EAC-TRS	Hot short disables E-TR-S	1.123	SAMA case FR-07b "Protect cables that would disable TR-S due to hot short" addresses this basic event.
10	FR1D	Fire in R1D - NE reactor building 471 elevation	1.111	A potential SAMA candidate is to install early detection in this compartment.
11	FW11	Fire in A HVAC room	1.109	A potential SAMA candidate is to install early detection in compartment RC-11.
12	HS-CIAV-MO20	CIA-V-20 failure due to hot short	1.091	This event affects the ability to open the SRVs. The FPSA conservatively does not credit the air accumulators installed at each of the SRVs. Therefore, this basic event is judged to not be a realistic contribution to risk. No SAMA candidates identified.
13	SW-----B----T3LL	SW pump SW-P-1B unavailable due to test or maintenance	1.088	SAMA case CW-07, "Add a service water pump", addresses this basic event.
14	HPS---------T3LL	HPCS pump HPCS-P-1 unavailable due to maintenance	1.078	SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
15	FP-FCP--------LL	Failure of fire control panel	1.07	This basic event represents failure of automatic fire suppression actuation. SAMA candidates identified in this table for installation of early fire detection address this basic event.
16	EACTRL-ASHE-W3D1	Loss of power from E-TR-S from the Ashe	1.069	This basic event represents the random LOOP supply from the Ashe substation. SAMA candidate AC/DC-27, "Install

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
		substation		permanent hardware changes that make it possible to establish 500 kV backfeed through the main step-up transformer," addresses this basic event.
17	EXF	Early extinguishment fails (power transformer)	1.064	This early extinguishment probability is empirical based on a review of the fire events database. New SAMA candidates identified in this table to install early fire detection systems in significant fire compartments will address this basic event.
18	FW03	Fire in cable chase	1.064	A potential SAMA candidate is to improve fire detection capability in the cable chase, such as an aspirating smoke detection system for early fire detection.
19	CIAHUMNV104BH 3-F	Operator fails to open manual block valve CIA-V-104B given fire event	1.063	This event affects the ability to open the SRVs. The FPSA conservatively does not credit the air accumulators installed at each of the SRVs. Therefore, this basic event is judged to not be a realistic contribution to risk. No SAMA candidates identified.
20	HS-RHRV-MO-23	RHR-V-23 fails due to hot short	1.062	This hot short fails suppression pool cooling train B. SAMA candidate CP-01, "Install an independent method of suppression pool cooling," addresses this basic event.
21	ECF	Early extinguishment fails (cabinet / electrical panel)	1.059	This early extinguishment probability is empirical based on a review of the fire events database. New SAMA candidates identified in this table to install early fire detection systems in significant fire compartments will address this basic event.
22	HS-RHRV-MO-6B	RHR-V-6B fails due to hot short	1.058	This hot short fails suppression pool cooling train B. SAMA candidate CP-01, "Install an independent method of suppression pool cooling," addresses this basic event.
23	FP-V-CLAPPERW2LL	Failure of deluge valve to open	1.057	This basic event results in failure of automatic suppression. New SAMA candidates for installation of early fire detection

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
				address this basic event.
24	HS-CIAV-MO30B	CIA-V-30B failure due to hot short	1.057	This event affects the ability to open the SRVs. The FPSA conservatively does not credit the air accumulators installed at each of the SRVs. Therefore, this basic event is judged to not be a realistic contribution to risk. No SAMA candidates identified.
25	FW08	Fire in division 2 switchgear room	1.051	A potential SAMA candidate is to improve fire detection capability in this compartment, such as an aspirating smoke detection system, for early fire detection.
26	EACENG-EDG2-S4D2	DG-2 fails to run	1.048	SAMA candidate AC/DC-10, "Provide an additional diesel generator," addresses DG-1 unavailability. Since DG-2 is more important to fire risk, an additional SAMA candidate to examine risk improvement for DG-2 will be considered.
27	FY01	Fire in transformer yard	1.045	This basic event represents impacts to the offsite power supply due to fire. SAMA candidate AC/DC-27, "Install permanent hardware changes that make it possible to establish 500 kV backfeed through the main step-up transformer," addresses this fire impact.
28	FW10	Fire occurs in main control room.	1.043	A potential SAMA candidate is to install early detection in the main control room.
29	ADSHUMN--T--H3-F	Operator fails to initiate depressurization given internal fire event	1.041	This human failure event involves the operator failing to discern that ADS should have initiated and did not or, having inhibited the ADS function, fails to manually initiate it when needed. No further refinements identified for this operator action. No further improvement of the operator action was identified. CGS has redundant automatic ADS functions. In accordance with EOPs the operator may temporarily delay the ADS

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
				function for 105 seconds by resetting the time delay or bypass of the ADS function through inhibit switches. Rigorous training and procedural controls exist to assure this key operator action is accomplished. Establishing an automatic function for resetting the ADS inhibit would remove important operator flexibility in diagnosing and implementing the EOPs and SAGs at the proper time for emergency depressurization.
30	EACENG-EDG3-S424	DG-3 fails to run for 24 hours	1.041	DG-3 provides power to HPCS. SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
31	PRAFN--1B---R3	Fan PRA-FN-1B fails to start on demand	1.041	This fan provides room cooling to SW train B. SAMA case CW-07, "Add a service water pump", addresses this basic event.
32	EACENG-EDG2-R3D2	DG-2 fails to start	1.035	SAMA candidate AC/DC-10, "Provide an additional diesel generator," addresses DG-1 unavailability. Since DG-2 is more important to fire risk, an additional SAMA candidate to examine risk improvement for DG-2 will be considered.
33	FW05	Fire in battery room 1	1.035	A potential SAMA candidate is to improve fire detection capability in this compartment, such as an aspirating smoke detection system, for early fire detection.
34	HS-RHRV-MO-16B	RHR-V-16B fails due to hot short	1.035	This hot short fails suppression pool cooling train B. SAMA candidate CP-01, "Install an independent method of suppression pool cooling," addresses this basic event.
35	HS-RHRV-MO-17B	RHR-V-17B fails due to hot short	1.035	This hot short fails suppression pool cooling train B. SAMA candidate CP-01, "Install an independent method of suppression pool cooling," addresses this basic event.

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
36	RHRV-MO--6B-O2LL	RHR-V-6B fails to remain closed	1.035	This random failure fails suppression pool cooling train B. SAMA candidate CP-01, "Install an independent method of suppression pool cooling," addresses this basic event.
37	CIAHUMNX-TIEH3-F	Operator fails to properly line up CAS crosstie manual valves when required	1.032	This is a potential human recovery event for opening the SRVs. The FPSA conservatively does not credit the air accumulators installed at each of the SRVs. Therefore, this basic event is judged to not be a realistic contribution to risk. No SAMA candidates identified.
38	E10W10	Fire not extinguished in less than 10 minutes	1.032	This early extinguishment probability is empirical based on a review of the fire events database. SAMA candidates identified in this table to install early fire detection systems in significant fire compartments will address this basic event.
39	FW02	Fire in cable spreading room	1.031	A potential SAMA candidate is to improve fire detection capability in this compartment, such as an aspirating smoke detection system, for early fire detection.
40	FP-SENSOR-----LL	Failure of fire detection sensor	1.029	This basic event results in failure of automatic suppression. New SAMA candidates for installation of early fire detection address this basic event.
41	HPSV-CH----5P5LL	HPCS-V-5 check valve fails to open	1.029	SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
42	FW13	Fire in emergency chiller room	1.028	A potential SAMA candidate is to improve fire detection capability in this compartment, such as an aspirating smoke detection system, for early fire detection.
43	HPSV-MO---23O2LL	HPCS-V-23 fails to remain closed	1.028	SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
44	EACEDG-2----T3D2	DG-2 out of service due to maintenance	1.025	SAMA candidate AC/DC-10, "Provide an additional diesel generator," addresses DG-1 unavailability. Since DG-2 is more important to fire risk, an additional SAMA candidate to examine risk improvement for DG-2 will be considered.
45	EACENG-EDG1-S4D1	DG-1 fails to run	1.024	SAMA candidate AC/DC-10, "Provide an additional diesel generator," addresses this basic event.
46	SW-P-MDSWP1BS4LB	SW-P-1B fails to run	1.024	SAMA case CW-07, "Add a service water pump", addresses this basic event.
47	E15W10	Fire not extinguished between 10 to 15 minutes	1.022	This early extinguishment probability is empirical based on a review of the fire events database. New SAMA candidates identified in this table to install early fire detection systems in significant fire compartments will address this basic event.
48	FT1A	Fire in turbine building west 441	1.022	A potential SAMA candidate is to install early detection in this compartment.
49	RCITDP-24HR1S4LL	RCIC pump RCIC-P-1 fails to run	1.022	Impacts RCIC operation. SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
50	DMAFN--31---R3	Fan DMA-FN-31 fails to start	1.021	Impacts DG-3 operation (HPCS diesel). SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event. ".
51	RRAFN--RFC04R3D3	Motor for fan RRA-FN-04 fails to start	1.021	Impacts HPCS operation. SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
52	NREAC6	Non-recovery of diesel in 6 hours	1.019	SAMA candidate AC/DC-10, "Provide an additional diesel generator," addresses this basic event.
53	EACEDG-1----T3D1	DG-1 out of service due to maintenance	1.018	SAMA candidate AC/DC-10, "Provide an additional diesel generator," addresses this basic event.

Table 5.d-1: Identification of Additional Potential SAMA Candidates from the FPSA				
Rank	Basic Event Name	Basic Event Description	RRW	Review of Existing and Potential SAMA Candidates
54	FT12	Fire in Turbine building south corridors	1.018	A potential SAMA candidate is to install early detection in this compartment.
55	EACENG-EDG1-R3D1	DG-1 fails to start	1.017	Impacts RCIC operation. SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
56	EACENG-EDG3-R3D3	DG-3 fails to start	1.017	DG-3 provides power to HPCS. SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
57	FW1A	Fire in radwaste building 437-foot elevation	1.017	A potential SAMA candidate is to install early detection in this compartment.
58	N24AVE-F	Average probability for non-recovery of AC power - fire	1.017	SAMA candidate AC/DC-27, "Install permanent hardware changes that make it possible to establish 500 kV backfeed through the main step-up transformer," addresses this fire impact.
59	RHR----A----T3LL	RHR train A unavailable due to test or maintenance	1.016	SAMA case CP-01 "Install an independent method of suppression pool cooling" addresses this basic event.
60	EACEDG-123FRC3LL	CCF of all three DGs to run	1.015	SAMA candidate AC/DC-28, "Reduce common cause failures between EDG-3 and EDG1/2," addresses this basic event.
61	HS-TSWV-MO-53A	TSW-V-53A fails due to hot short	1.015	Loss of TSW impacts RFW operation. SAMA candidate CC-01, "Install an independent active or passive high pressure injection system," addresses this basic event.
62	RHRP-MD---2BS4LL	RHR-P-2B fails to run	1.015	This random failure fails suppression pool cooling train B. SAMA candidate CP-01, "Install an independent method of suppression pool cooling," addresses this basic event.

NRC Request:

e. ER Section E.9.3 discusses significant contributors to the top 100 Level 1 PSA cutsets. Table E.9-1 shows that there are a number of operator errors and non-recovery actions that occur in this listing of dominant cutsets. Given that there are operator errors that repeat in a large number of cutsets (e.g. "Operator Fails to Initiate Depressurization during Non-ATWS Event") and that operator errors often have relatively high failure probabilities explain why no operator errors were identified as the basis for a plant-specific SAMA (e.g., an improvement to a specific procedure). Explain how human errors identified in the PSA were considered in the candidate SAMA identification process.

Energy Northwest Response to 5.e:

Operator errors, along with common cause failures, typically contribute to dominant cutsets in PSAs. This was true in the CGS PSA model. Significant HRA model improvements and procedure enhancements were made in the incorporation of 2004 peer review F&Os. Review of the Revision 6.2 important HEPs and industry procedural enhancements (see CC-04 through CC-07, CC-10, CC-11, CC-16, CC-17, CC-19 and others) were already implemented at CGS. CGS has placed considerable emphasis on procedure use and enhancement to improve operator response. Other SAMA candidates associated with procedure enhancement were considered for further evaluation (AC/DC-23 & CC-21). The general assessment is that CGS's plant procedures and training have had significant enhancement and the review did not identify additional inherent weaknesses that could be enhanced to improve operator actions.

Some of the important HEP basic events in Revision 6.2 have had procedural and operator aid improvements, but these improvements had not been incorporated in the PSA used to perform the SAMA analysis. Table 5.e-1 lists important HEPs that have had improvements in either their risk modeling (specific HEP derivation by use of the EPRI HRA Calculator) or in procedure enhancements and these improvements have been incorporated in the recently upgraded model.

The sensitivity comparison confirms that additional SAMA candidates to further improve operator performance would not be found cost effective primarily due to their low risk benefit.

One potential SAMA candidate was identified for analysis with PSA Rev. 7.1 to increase the operator awareness of time critical and high risk important operator actions within the operations procedures. Although this SAMA is not directed to

a specific important HEP basic event, it could improve the operators' performance for those important HEPs. A sensitivity study will be performed (PSA Rev. 7.1) by decreasing significant HEPs by an appropriate factor in the upgraded model to assess if sufficient risk benefit results that a general training and procedural update associated with time critical and high risk important operator actions would be cost beneficial. This is similar to screened candidate OT-05 in Table E10-1 as very low benefit. This was also identified as an industry cost beneficial SAMA and included as "Criterion C – Considered for Further Evaluation" in the response to RAI 5.a. This sensitivity SAMA evaluation will assess its disposition.

Table 5.e-1: Sensitivity Evaluation of Important Human Error Basic Events						
HEP Basic Event	Description	HEP Rev. 6.2	HEP Rev. 7.1	RRW Rev 6.2	RRW Rev 7.1	Discussion
ADSHUMNSTARTH3LT	Operator fails to initiate depressurization during non-ATWS event	2.8E-04	1.47E-4	1.207	1.039	This human failure event involves the operator failing to discern that ADS should have initiated and did not or having disabled the ADS function fails to manually initiate it when needed. No further refinements identified for this operator action. CGS has redundant automatic ADS functions. In accordance with EOPs the operator may temporarily delay the ADS function for 105 seconds by resetting the time delay or bypass of the ADS function through inhibit switches. Rigorous training and procedural controls exist to assure this key operator action is accomplished. Establishing an automatic function for resetting the ADS inhibit would remove important operator flexibility in diagnosing and implementing the EOPs and SAGs at the proper time for emergency depressurization.

Table 5.e-1: Sensitivity Evaluation of Important Human Error Basic Events						
HEP Basic Event	Description	HEP Rev. 6.2	HEP Rev. 7.1	RRW Rev 6.2	RRW Rev 7.1	Discussion
RCIHUMNOVRIDE3LL	Operator fails to override false RCIC low discharge pressure signal	1.0E+00	3.31E-3	1.106	Negl.	Given a spurious low discharge pressure alarm for RCIC, the system is expected to trip. Procedures allow operators to override the signal to attempt to restore RCIC. This action is proceduralized and an HEP was developed for PSA Rev. 7.1. RRW importance is negligible.
WMAHUMNALTCCINLL	Operator fails to provide alternate ventilation given a loss of division 1 and division 2 switchgear room cooling initiating event	1.0E-02	n/a	1.099	n/a	For PSA Rev. 7.1, an improved procedure based on thermal dynamic calculations for fan placement and timing was developed. Dual loss of switchgear room cooling was screened as an initiating event based on ample time for operators to detect and correct the issue prior to the time that plant shutdown would be required.
XDPHUMN-INJ-AHR-	Operator fails to initiate ADS and control HPCS/RCIC	9.0E-06	1.25E-05	1.017	1.013	The HEP for this dependent human failure event is low. No further refinements are identified.

Table 5.e-1: Sensitivity Evaluation of Important Human Error Basic Events						
HEP Basic Event	Description	HEP Rev. 6.2	HEP Rev. 7.1	RRW Rev 6.2	RRW Rev 7.1	Discussion
XDPHUMN-INJ-HRFA	Operator fails to initiate ADS and control RFW / HPCS / RCIC	1.0E-06	4.8E-06	1.01	1.003	The HEP for this dependent human failure event is low. No further refinements are identified.
XDPHUMN-INJ-RA--	Operator fails to initiate ADS and control RCIC	1.5E-05	2.5E-05	1.009	1.010	The HEP for this dependent human failure event is low. No further refinements are identified.
XDPHUMN-DHR-VSX-	Operator fails to initiate suppression pool cooling, containment venting, and fails to initiate ADS and control HPCS/RCIC	5.0E-07	n/a	1.008	n/a	The HEP for this dependent human failure event is low. No further refinements are identified. For PSA Rev. 7.1, this human failure event combination was identified to not be significant and no dependent human failure event was modeled.
WMAHUMNALTCCF3LL	Operator fails to supply alternate ventilation given a loss of division 1 and 2 switchgear room cooling post-initiator	1.0E+00	7.1E-03	1.007	1.000	For PSA Rev. 7.1, an improved procedure based on thermal dynamic calculations for fan placement and timing was developed. Event importance reduced significantly.

NRC Request:

f. ER Section E.9.5 identifies the major contributors to each release category but does not identify SAMAs to address the major contributors. Clarify how these specific risk contributors were considered to identify potential candidate SAMAs. Identify and evaluate SAMAs to address all of the major contributors having an RRW benefit value greater than the minimum cost of a procedure change at CGS.

Energy Northwest Response to 5.f:

The dominant risk contributor to the Level 2 analysis is from the large early release category. For the LEN category, major contributors are listed in Section E.9.5, along with SAMA candidates to reduce the frequency of this release category. See also the response to RAI 2.f.

For the other release categories, LLN and LLS, Section E.9.5 provides the major contributors and discusses in general terms the SAMAs evaluated for these contributors. Specific SAMA candidates that were examined that apply to these two release categories will be provided in the updated Section E.9.5 in Amendment 10 as follows:

LLN

- Loss of all ECCS injection: Candidates CC-01 and CC-02 (high pressure (HP) injection)
- Loss of suppression pool (SP) cooling : CP-01 (additional SP cooling train)
- Long term SBO: Numerous AC/DC SAMA cases were examined including AC/DC-1, AC/DC-10, AC/DC-15, and AC/DC-28

LLS

- Loss of HP injection: CC-01 and CC-02 (HP injection)
- RPV Rupture (no SAMA candidates)

NRC Request:

g. The source of SAMA AC/DC-29 is not identified in ER Table E.9-3. In light of the fact that AC/DC-29 was one of the few plant-specific SAMAs identified, specify how SAMA AC/DC-29 was identified.

Energy Northwest Response to 5.g:

SAMA candidate AC/DC-29 was developed based on a review of the significant Level 1 cutsets provided in Table E.9-1. The review identified several significant cutsets that included LOOP in combination with the failure of all three DGs to start or run. A reference to Table E.9-1 was added to the source column of Table E.9-3 in Amendment 10 for AC/DC-29.

NRC Request:

h. ER Section E.9.2, page E-66, states that a cost-benefit analysis was performed on increasing the capacity of the 230 kV/115 kV plant bus transfer and found this modification to not be cost effective. Provide a summary of the scope and results of this analysis and clarify the criteria used to determine the cost effectiveness of the modification.

Energy Northwest Response to 5.h:

The scope of the modification evaluated was limited to increasing the capacity of the CGS 230 kV startup transformer. This offsite source is the primary offsite power source and its loading has less margin than the 115 kV transformer. The evaluation of cost effectiveness therefore focused on the 230 kV transformer.

The 230 kV transformer is limited in supplying both Division 1 and Division 2 ECCS pumps simultaneously. The increased capacity would allow higher reliability from the probability of an under-voltage condition during ECCS pump starting and provide additional capacity for other non-safety related loads that could provide additional risk benefit. The higher capacity could also be used to reduce the time delay between ECCS pump starting times. The evaluation calculated a decrease of 7.0E-7/yr of baseline CDF using Rev. 1 of the IPE. The analysis assumed a benefit of $250,000 for each decrease of 1 E-06/yr in CDF. A higher capacity transformer's cost and installation was estimated at $2,000,000 and thus deemed not cost effective.

AC/DC-27, which proposed adding additional equipment for making the 500 kV backfeed available, represents a similar SAMA. The benefits and costs are similar. AC/DC-27 was found to be not cost effective.

NRC Request:

i. ER Section E.9.2 presents the status of improvements identified by Revision 1 of the IPE, with the exception of the suggestion to modify back-up air supply to the mainstream isolation valves (MSIVs) and containment vent valves. Section 6.2 of Revision 1 of the IPE identifies that potential improvement as marginally cost effective but notes that the improvement could increase in importance as the other improvements were implemented. Given that other improvements have been implemented and that the importance of the closure of these valves is still indicated by the current PSA, provide a cost-benefit analysis of a SAMA to modify to provide redundant air supply to the MSIVs and containment event valves.

Energy Northwest Response to 5.i:

The IPE stated the following:

"CN Supply to MSIVs and Vent Valves - A hardware modification of the air supply to the inboard MSIVs and the containment vent valves for backup from the containment nitrogen system was investigated. The modification would improve long term decay heat removal by providing redundant air supply to the valve's solenoids. As discussed in Section 3.4.3, the total contribution to CDF from loss of decay heat removal function is 1.4E-6 per year. The hardware modification costs are marginally cost effective. It is not recommended for further plant evaluation at this time but may increase in importance as other recommendations are instituted."

The MSIVs provide two primary functions. MSIV closure is the safety related function for containment isolation and inventory control. MSIV (re)opening is a risk improvement function. Operating air is supplied to the outboard valves from the plant air system and to the inboard valves from the containment instrument system (nitrogen). An air accumulator between the control valve and a check valve provides backup operating air. The outboard MSIVs will close on spring force or air cylinder pressure; the inboard valves require spring force and air pressure to close. The function identified in the IPE was for additional gas supply to support maintaining the MSIVs open or reopening to establish a DHR capability through the condenser.

The CGS current design is that the primary supply to the inboard MSIV is from the containment nitrogen (CN) supply. This is the preferred supply to limit de-inerting the containment atmosphere through gas leakage if air was the normal supply. The CN storage tank is an 11,000 gallon liquid nitrogen tank. There are additional backup supplies to the nitrogen supply for the inboard MSIVs. The additional backup supplies are from the control and

service air system. This system consists of three control air compressors and one service air compressor that can supply back up air pressure to the CN supply. The control and service air system provides air to the outboard MSIVs.

The change in CDF by making gas supply to the MSIVs perfect is negligible (RRW = 1.000). Thus, the addition of an air or nitrogen supply would provide a low risk benefit and was appropriately screened.

The air supply to the containment vent valves is from the control and service air system (4 air compressors). The improvement identified by the IPE regarding containment vent was to provide operators additional capability to open the containment vent valves for mitigating potential containment over-pressurization and providing removal of decay heat. This plant improvement was accomplished by creating a procedure that uses portable nitrogen bottle(s) to manually open the vent valves. However, the RRW for the air supply to the containment vent valves is 1.0002. The PSA was not updated to incorporate this manual function due to its low risk significance. The addition of another air or nitrogen supply would provide a low risk benefit and was appropriately screened.

NRC Request:

j. ER Section E.3.2.4, in the discussion of seismic-related improvements, states that a cost-benefit analysis was performed on strengthening the motor control centers (MCC) base connections and found this modification to not be cost effective. Provide a summary of the scope and results of this analysis and clarify the criteria used to determine the cost effectiveness of the modification. Justify why this improvement should not be addressed as a SAMA in light of more recent seismic hazard curve data (see RAI 3.a).

Energy Northwest Response to 5.j:

The recent seismic hazards curves in RAI 3.a have been shown to be consistent with the CGS seismic hazard curves used for the SPSA. No change to the CGS seismic hazard curve is warranted. Thus, the fragility of the MCCs would not change. However, a review of the IPE for the change to strengthen the MCCs base was performed to assess its adequacy using the criteria of NEI 05-01.

The review of improvements for the IPEEE identified that strengthening the seismic ruggedness of two MCCs (MCC-7F and MCC-8F) would provide

improvement to seismic risk. The cost benefit analysis provided was different from that appropriate for SAMA analysis.
A sensitivity study is planned with an updated cost estimate and the updated Revision 1 of CGS SPSA model, which has been integrated with the upgraded CGS Internal Events PSA Revision 7.1.

NRC Request:

k. ER Section 9.2 under the discussion of IPEEE insights indicates that two seismic SAMA candidates were evaluated. Tables E.11-2 through E.11-5 appear to provide Phase II evaluation results for only one seismic SAMA, SAMA SR-03 (which was identified from the NEI 05-01 generic SAMA list). Clarify this discrepancy.

Energy Northwest Response to 5.k:

SAMA candidate SR-01's screening listing in Table E.10-1 was "considered for further evaluation." However, also in Table E.10-1, the Basis for Screening/Modification Enhancements provided a qualitative basis for screening the potential enhancement. The screening criterion should have listed the potential SAMA candidate as Criterion E "very low benefit." The screening criterion for SR-01 in Table E.10-1 was updated and Section E.9.2 in Amendment 10 was corrected to state one seismic SAMA candidate was chosen for cost-benefit analysis.

NRC Request:

l. ER Section 9.2 under the discussion of IPEEE insights states "the dominant fire sequences render containment venting, power conversion system (PCS) and one train of RHR or service water unavailable." SAMAs FR-07a and FR-07b were identified to address protecting the containment vent valve cables and transformer E-TR-S cables, respectively, from fires. In light of the fact that these two SAMAs are cost-beneficial at a 3 percent discount rate, provide a cost-benefit analysis of a SAMA(s) to protect RHR and service water cables from fires.

Energy Northwest Response to 5.l:

A cost-benefit analysis of a SAMA candidate to protect RHR and SW cables from fires was performed. This SAMA candidate examined replacing specific cables with more fire resistant cables to reduce the failure from fire. Although CGS electrical cabling is protected from fire to manually shutdown in the RHR alternate shutdown mode (Appendix R), the proposed SAMA would require additional protection from MSOs in auto initiation circuits of RHR and SW.

A considerable amount of cabling is involved. The total benefits for the baseline and sensitivity case for the 3% discount rate for this SAMA candidate are $523,791 and $794,279, respectively. The implementation cost for this SAMA candidate was estimated to be $1,250,000. Based on this implementation cost, the SAMA candidate is not cost-effective to implement at CGS.

NRC Request:

m. ER Table E.10-1 screens outs several candidate SAMAs using the code, Criterion E (Very Low Benefit). Provide the following relative to these screened SAMAs:

i) It is not clear how the risk of particular failures can be determined to be insignificant without an importance analysis that includes specific consideration of related basic events. By way of illustration, in Table E.10-1 for SAMA CC-09, opening a SRV is stated to be very reliable and not a significant contributor to risk. Clarify the basis for stating that opening a SRV is not a significant contributor to risk, clarify like-kind statements in Table E.10-1, and provide RRW to support the "very low benefit" conclusion.

ii) The basis for screening SAMA FW-04 appears to presume that feedwater unavailability is more sensitive to loss of flow from the condensate and condensate booster pumps than from independent or common cause failures of the feedwater pumps themselves, such that additional redundancy of the latter would not significantly improve FW availability. Clarify if this understanding is correct and provide the RRW for the different systems (i.e., condensate booster pumps and feedwater pumps).

Energy Northwest Response to 5.m:

i) As part of evaluating CC-09, SRVs failing to open did not appear in any of the dominant cutsets as presented in Table E.9-1 and the SRVs fail to open did not have a RRW value of 1.005 or higher. Therefore, any improvements to ensure SRVs opening were considered to be of low benefit.

Several factors were used to conclude that a potential SAMA candidate was of very low benefit. Both qualitative (e.g., dependencies) and quantitative (e.g., cutsets, importance values) arguments were applied. The bases for designating something as very low benefit were provided in the description of the qualitative screening table. It is recognized that

those SAMA candidates that were designated as very low benefit based on quantitative risk arguments did not always have quantitative values presented in this table. For those cases, quantitative information (e.g., RRW) has now been provided in the updated Table E.10-1 in Amendment 10.

ii) The fault tree analysis provided in the RFW system notebook calculated that the top 79% contributors to RFW unavailability were factors other than the RFW pumps. These failures included such items such as operator error, electrical power and breaker failures and motor operator valve failures. RFW is also dependent on the condensate pumps and components for success, but it was not meant to imply that they were larger contributors to RFW unavailability. It was concluded that adding an additional motor-driven RFW pump would add little benefit versus the cost incurred. However, the importance of RFW has increased in the upgraded PSA Revision 7.1. Therefore, it is appropriate to reevaluate SAMA FW-04 as part of the sensitivity study identified in response to RAI 1.a to verify its cost-beneficial disposition.

NRC Request:

n. The internal events, fire, and seismic CDF, annual dose, and annual property loss results in ER Table E.11-1 through E.11-4 are shown to be 0.00E+00 for many cases. Insufficient information is provided to ascertain whether these results are truly 0.00E+00 or are negligible. For all instances in which these results are shown to be 0.00E+00, clarify whether the results are actually zero (SAMA has no impact) or whether the results are negligible (SAMA has a negligible impact). In addition, justify the assignment of zero risk reduction for the following Cases: AC/DC-01 (Fire), CC-20 (all), CB-01 (all), and AT-14 (Fire and Seismic).

Energy Northwest Response to 5.n:

a) Zero Values

The delta-CDF and delta-LERF values in Table E.11-1 were calculated from CDF and LERF results recorded to four significant digits. For example, the internal events risk reduction delta CDF for AC/DC-01 was computed as follows:

Risk reduction CDF – Base CDF = Risk reduction delta CDF
4.769E-6/rx-year - 4.512E-6/rx-year = 2.6E-7

The internal events risk reduction LERF is:

Risk reduction LERF – Base LERF = Risk reduction delta LERF
6.530E-7/rx-year – 6.530E-7/rx-year = 0.00E+00

Thus, the 0.00E+00 values in Table E.11-1 are known to be zero in nearly every instance, with maximum round-off error on the order of 4E-10/rx-year for CDF, and 4E-11/rx-year for LERF.

In some cases the 0.00E+00 values indicate negligible impact. For example, for CB-01, the internal events risk reduction delta CDF is 1.0E-09, which was judged to be negligible and reported as 0.00E+00 in Table E.11-1. For case SR-03, the seismic risk reduction delta CDF is 1.0E-09, which was judged to be negligible and reported as 0.00E+00 in Table E.11-1

The information in Table E.11-1 is then used as input to the calculations performed in Tables E.11-2 through E.11-4. If the delta-CDF is zero in Table E.11-1, then the reduction in CDF in the subsequent tables is also zero. The values as provided to E.11-1 flow to the corresponding cases in the subsequent tables.

b) <u>Zero Risk Reduction assignment justification for Cases AC/DC-01 (fire), CC-20 (all), CB-01 (all), and AT-14 (fire and seismic)</u>

AC/DC-01: This SAMA would add battery capacity to maintain RCIC operating for up to 10 hours during an SBO thereby increasing the time for recovery of offsite power. LOOP is caused by fire damage to applicable cables and no recovery potential is postulated (the probabilities for non-recovery of offsite power are 1.0 for the FPSA). Therefore, there is no risk reduction from providing additional DC power capacity for maintaining RCIC operating for additional time until offsite power can be recovered. The fire damage is assumed significant enough that restoration would not be possible in the near time frame and onsite power systems are the remaining means of power to critical busses. Future upgrade of the FPSA may identify certain cables that may be restorable or breakers that tripped on excessive temperature but was not damaged by the fire directly that could be reset. However, the existing FPSA model is not capable of generating realistic benefit values for offsite power recovery.

CC-20 (all): This SAMA would improve ECCS strainers or replace containment insulation to minimize strainer clogging. Each of the redundant suction strainers is modeled in PSA Rev 6.2 as independent

from the other strainers, and thus, this SAMA case registers no delta CDF. This is an area of model incompleteness.

A sensitivity evaluation was performed to examine the impact to the CGS risk profile of modeling common cause failures of the ECCS pump suction strainers. The need to examine this possible modeling change was identified by the 2004 peer review as a Category C non-significant F&O. The increase in total CDF (internal events, fire and seismic, combined) for the base PSA average maintenance model was 3E-9/yr, which was reported as 0.00E+00 but is better characterized as a negligible impact.

CB-01 (all): This SAMA would improve the ability to monitor and respond to an ISLOCA. The ISLOCA contribution to PSA results (internal events) was removed from the solution to maximize the benefits for internal events (no ISLOCA initiating event). The risk delta CDF for internal events was computed to be 1.0E-9/rx-year, which is a negligible contributor.

The FPSA does not model the potential for fire-induced ISLOCA, however the risk contribution is judged to be small. ISLOCA in the shutdown cooling suction line (RHR-V-8 and RHR-V-9 in series) is of low likelihood. Valve RHR-V-9 is maintained in the closed position during normal plant operation with power removed from the motor via a protected isolation switch maintained in the "ISOLATE" position. Spurious control and power signals resulting from hot shorts cannot cause the valve motor to energize. The de-energized (isolated) power feeder has been routed in a grounded steel conduit to protect it against external three-phase hot shorts. For other pathways, a hot short plus random failure of a check valve is required to produce an ISLOCA. This area of incompleteness uncertainty is judged to be negligible contributors to the FPSA results.

Seismic damage state 41 includes seismic-induced failures of key systems (e.g., DC power, RHR heat exchangers, control room main panels, etc.). Seismic damage state 42 is comprised of seismic-induced failures of the RPV and key buildings, any one of which is modeled as leading to core damage. These seismic damage states include potential ISLOCA, but there is no way to differentiate this potential from other serious impacts that contribute to core damage.

AT-14: This SAMA would add diversity between the two standby liquid control (SLC) explosives valves to increase the reliability of SLC.

Fire-ATWS is not modeled in the FPSA, on the basis of low risk significance (Table 2-1 of NUREG/CR-6850); thus, there is very little risk reduction potential for fire.

Seismic-ATWS - The dominant seismic damage state sequence is SDS40 (7.93E-9). It is an unmitigated (i.e., fails to scram and fails to mitigate; SLC fails and ADS inhibit fails) seismic-induced ATWS scenario. This damage state leads directly to core damage. The diversification of SLC explosive valves would not enhance the mitigation of this sequence. To improve mitigation, a significant increase in seismic ruggedness in the diverse SLC valve and its piping would be required. Since the SLC valves take suction from a common SLC storage tank and discharge to a common injection line, significant improvement in seismic ruggedness is not practicable. Table E.11-1 correctly reported this as 0.00E+00.

NRC Request:

6) Provide the following with regard to the Phase II cost-benefit evaluations:

   a. On ER page 3-1 it is stated that the net and gross electrical power outputs are 1,190 and 1,230 MWe, respectively. However, on page E-62 a CGS rated electrical power of 1107 MWe is used in estimating replacement power costs. Provide the rationale for using 1107 MWe in estimating the replacement power cost used in the SAMA analysis. In addition, provide an assessment of the impact on the SAMA analysis of using 1,190 MWe in estimating power replacement cost.

Energy Northwest Response to 6.a:

The CGS estimate for the replacement power costs was based on an electrical output power of 1,107 MWe. This was determined using the net electrical power output of 1,190 MWe at a 93% capacity factor.

A sensitivity case using 100% net electrical power was performed, as requested, to assess the impact on the SAMA candidates in Table E.11-7. The sensitivity case used an electrical power output of 1,190 MWe to estimate replacement power costs. Table 6.a-1 provides the results of the sensitivity case. The sensitivity analysis did not result in any of the SAMA candidates being cost beneficial to implement at CGS. Therefore, using an electrical power output of 1,190 MWe to determine replacement power cost does not impact the conclusions of the SAMA analysis provided in Section E.13.

Table 6.a-1: Cost-Benefit Results for Increased Replacement Power Cost Sensitivity Case			
SAMA ID	**Increased Power Sensitivity Case Total Benefit**	**Cost Estimate (2008)**	**Conclusion**
AC/DC-01	$38,345	$1,799,200	Not Cost Effective
AC/DC-10	$261,441	$10,816,000	Not Cost Effective
AC/DC-23	$21,121	$375,000	Not Cost Effective
AC/DC-27	$311,951	$1,700,000	Not Cost Effective
AC/DC-28	$75,126	$100,000	Not Cost Effective
AC/DC-29	$157,150	$4,200,000	Not Cost Effective
AT-05	$5,680	$800,000	Not Cost Effective
AT-07	$0	$1,124,864	Not Cost Effective
AT-13	$189	$660,000	Not Cost Effective
AT-14	$379	$370,000	Not Cost Effective
CB-01	$0	$5,600,000	Not Cost Effective
CB-03	$0	$400,000	Not Cost Effective
CB-08	$0	$20,000	Not Cost Effective
CB-09	$0	$30,000	Not Cost Effective
CC-01	$916,502	$29,120,000	Not Cost Effective
CC-02	$916,502	$5,200,000	Not Cost Effective
CC-03b	$55,569	$160,000	Not Cost Effective
CC-20	$0	$10,000,000	Not Cost Effective
CP-01	$566,703	$6,000,000	Not Cost Effective
CW-02	$26,060	$650,000	Not Cost Effective
CW-03	$110,665	$1,124,864	Not Cost Effective
CW-04	$110,665	$675,000	Not Cost Effective
CW-07	$185,930	$6,136,000	Not Cost Effective
FR-03	$221,965	$2,000,000	Not Cost Effective
FR-07a	$353,366	$400,000	Not Cost Effective
FR-07b	$80,081	$100,000	Not Cost Effective
HV-02	$220,057	$480,000	Not Cost Effective
SR-03	$0	$980,000	Not Cost Effective

NRC Request:

b. ER Table E.10-1 identifies many Phase I SAMAs that were subsumed into other Phase I SAMAs and not analyzed in the Phase II evaluation. However, the SAMAs that were subsumed may have a lower cost of implementation than the SAMA actually evaluated (i.e., subsumed SAMA AC/DC-03 appears to have a lower cost than SAMA AC/DC-01 which was evaluated). Provide a cost estimate and Phase II cost-benefit evaluation for all subsumed SAMAs or, alternatively, justify that the implementation cost of the subsumed SAMA is higher than the implementation cost of the evaluated SAMA.

Energy Northwest Response to 6.b:

The SAMA candidates subsumed in Phase I (AC/DC-02, AC/DC-03, AC/DC-15 and AC/DC-16) have a lower cost of implementation than the SAMA candidates evaluated in Phase II. An analysis was performed to assess the cost-benefit of the subsumed candidates. The total benefit from internal events, fire, and seismic was derived and compared to the cost of implementation. Table 6.b-1 provides the results of the cost-benefit evaluation. None of the subsumed SAMA candidates are cost-beneficial to implement at CGS. Therefore, the conclusion in Section E.13 remains applicable.

The screening criteria in Table E.10-1 stated SAMA candidates CB-03, CB-08, and CB-09 were subsumed by candidate CB-01. However, these SAMA candidates were evaluated in Phase II of the SAMA analysis. The screening criteria for these SAMA candidates were modified from "subsumed" to "consider for further evaluation" for clarification. Table E.10-1 was updated in Amendment 10 for clarification.

Table 6.b-1: Final Results of the Cost-Benefit Evaluation for Subsumed SAMA Candidates

SAMA ID	Modification	Analysis Case	Estimated Benefit	2008 Estimated Cost	Conclusion
AC/DC-02	Replace lead-acid batteries with fuel cells.	Case 01	$37,451	$1,040,000	Not Cost Effective
AC/DC-03	Add a portable, diesel-driven battery charger to existing DC system.	Case 01	$37,451	$500,000	Not Cost Effective
AC/DC-15	Install a gas turbine generator.	Case 02	$251,584	$2,080,000	Not Cost Effective
AC/DC-16	Install tornado protection on gas turbine generator.	Case 02	$251,584	>$2,080,000	Not Cost Effective

NRC Request:

c. With regard to the modeling assumptions for each SAMA described in ER Table E.11-1, provide the following:

   i) SAMA AC/DC-01 – time available to recover offsite power in the baseline PSA.
   ii) SAMAs AC-DC-10, CW-02, CW-03, CW-07, HV-02, AT-05, SR-03 – description of model changes in layman terms.
   iii) SAMA AC/DC-27 – description of the changes made to the PSA fault tree to model the 500 kV power source.
   iv) SAMA CC-03b – meaning of parenthetical phrase "see RCIC FTR tab"
   v) SAMAs FR-07a, FR-07b – description of the specific changes made to the fire model, in layman terms.

Energy Northwest Response to 6.c:

The approach to modeling the benefit of the SAMA cases was to use a conservative methodology generally overestimating the benefit by the model change. The following description of model changes provides examples of this approach.

i) SAMA AC/DC-01 – time available to recover offsite power in the baseline PSA.

For SBO cases, this SAMA extends the time of RCIC operation before battery depletion. It is assumed that RCIC can successfully operate for 10 hours, at which time either offsite or onsite power must be recovered. For LOOP, the PSA model change allows 10 hours for recovery of offsite power with RCIC operating. Specifically, the period for offsite / onsite recovery of power was extended to ten hours during SBO when RCIC successfully starts and runs on DC power.

ii) SAMAs AC-DC-10, CW-02, CW-03, CW-07, HV-02, AT-05, SR-03 – description of model changes in layman terms.

- SAMA Case AC/DC-10 examined the provision of an additional DG that could be aligned to either Division 1 or Division 2 4.16 kV bus. The DG would differ in design from DG-1 and DG-2 to minimize the likelihood of DG common cause failure events. The model change consisted of modifying the PSA model to make DG-1 perfectly reliable to start and run. DG-1 was selected due to the RCIC dependency on DG-1 (Division 1 provides charging power for batteries which provide

control power to the RCIC system). This is judged to provide maximum benefit for the evaluation.

- SAMA Case CW-02 adds redundant DC control power for pumps. The fault tree structure for failure of control power was set to zero (guaranteed success) for ECCS pumps. DC power dependencies for RCIC were retained in the model, as there is little risk benefit from such a modification (unavailability of the pump itself dominates the unavailability for the RCIC system).

- SAMA Case CW-07 adds an additional standby SW pump. The model changes involved making one train of SW perfectly reliable.

- SAMA Case HV-02 provides a redundant train or means of switchgear room ventilation. The model change involved removing the switchgear dependencies on HVAC and eliminating the loss of HVAC initiating events. Additionally, accident sequences for the loss of HVAC initiating events were removed from the model solution.

- SAMA Case AT-05 adds an independent boron injection system. The model change involved making SLC perfectly reliable to start and run. For the Internal Events PSA, the C(3) functional equations were set to a low value by replacing these equations with a single basic event, NULL, which has a value of 1E-8. For seismic, seismic damage state 40 (ATWS) was set to zero by removing the seismic damage state 40 sequences from the quantification. There was no change to the FPSA model. ATWS sequences are not modeled by the FPSA based on low risk contribution.

- SAMA Case SR-03 examines safety related CST availability given seismic events. The model change credits the availability of the CST during seismic events.

iii) SAMA AC/DC-27 – description of the changes made to the PSA fault tree to model the 500 kV power source.

SAMA AC/DC-27 examines installing permanent hardware changes that make it possible to quickly establish 500 kV backfeed through the main step-up transformer to the normal auxiliary transformers. For internal events and fire, an unavailability of 1E-2 was assumed for the 500 kV backfeed basic event. This value is assumed to account for potential correlated unavailability of the 500 kV source when the 230 kV and 115 kV credited offsite power lines are unavailable, as well as the HEP to align backfeed (which is a small contribution relative to the assumed

unavailability of power). The backfeed was assumed to be free from the effects of fire for the FPSA quantification.

Backfeed was assumed to be unavailable for seismic analysis due to potential extensive losses of offsite power (implies that the 500 kV backfeed and the 230 kV and 115 kV transmission line towers or terminal insulators fragility at CGS are common cause due to seismic).

iv) SAMA CC-03b – meaning of parenthetical phrase "see RCIC FTR tab."

The "see RCIC FTR tab" parenthetical phrase is a reference to a portion of the engineering calculation developed to support the SAMA risk quantification. In the context of the report, the phrase is ambiguous. This calculation documents that the various RCIC Failure to Run unavailabilities were reduced by a factor of three in order to perform the quantification of SAMA Case CC-03b.

v) SAMAs FR-07a, FR-07b – description of the specific changes made to the fire model, in layman terms,

For SAMAs FR-07a, FR-07b, the following specific changes were made to the fire model:

a) FR-07a evaluated improvement of the fire resistance of cables to the containment vent valves. Fire damage to cables that support vent operation cause a failure of the containment venting function with probability 1.0 (containment vent requires both Division 1 and Division 2 control power to the vent valves, which are in series, and therefore fire damage that causes an open circuit of the control cables for a containment vent valve fails the vent pathway). To approximate the risk benefit from protecting cables for containment vent (valves, containment air and power supplies), fire-related failures of containment vent were set to zero.

b) FR-07b examines improvement of fire resistance of cables susceptible to disabling the TR-S transformer due to hot short. TR-S provides 230 kV offsite power from the Ashe substation. The hot short probability was reduced to zero for the TR-S transformer.

NRC Request:

d. ER Section E.6 and Table E.11-6 state that plant personnel developed CGS-specific implementation cost estimates for many of the SAMAs. Provide a description of: the process Energy Northwest used to develop the SAMA implementation costs and the level of detail used to develop the cost estimates (i.e., the general cost categories considered).

Energy Northwest Response to 6.d:

The process Energy Northwest used to develop the SAMA implementation costs consisted of the following:

(a) A small team of three Energy Northwest and consultant personnel with over 50 years of experience at CGS were established as a core estimating team. This team has a broad knowledge of CGS and a diverse background of over 90 years collective in the nuclear industry in the areas of electrical and mechanical engineering, field engineering, design engineering, construction management, operations and maintenance support, licensing, and PSA. Relying on their knowledge of the plant and plant procedures, the implementation costs were estimated. The core team consulted plant experts in their area of expertise and solicited conceptualization of the SAMA and reasonable estimates of costs through an interview type of process. This included areas such as fire protection, operations and maintenance procedures, operations, training, design engineering, and system engineering. The team evaluated the estimated costs associated with material, labor, engineering, licensing, training, procedures, and surveillance testing for the majority of the SAMA candidates. This process produced a reasonable estimate of the costs.

(b) For some SAMA candidates that had existing implementation costs from published documents (e.g., other completed SAMA analyses), the team consulted those estimates.

(c) For some complicated SAMAs, the estimated implementation costs were sufficiently greater than the maximum benefit such that it was not necessary to perform a detailed cost estimate and to do so would have a high level of uncertainty as a more detail cost estimate would require a higher level of conceptualized design.

The core team's focus was to establish reasonable but optimistic cost estimates that would underestimate the actual cost of implementation in order

to ensure that the estimates used in the cost benefit evaluation were conservative (i.e., under estimates the cost and over estimates the benefit).

These cost estimates were then used to calculate the cost benefit. After reviewing the results, the team did not perform an additional cost evaluation to add costs for such items as replacement power, recurring maintenance, training, or periodic replacement costs. The team believes that the three cost effective SAMAs listed in Table E.12-1 could be enhanced with additional cost refinement, but elected to accept the results and include the cost effective SAMAs for consideration of implementation. None of these SAMAs are age-related in nature.

For further information and details associated with specific cost estimate, please see the responses to RAIs 6.e and 6.f.

NRC Request:

e. ER Section E.11.2 also states that the SAMA implementation cost estimates accounted for inflation when using estimates from other SAMA analyses. Clarify how other cost factors were treated in these estimates and in the development of the site-specific cost estimates. Specifically address 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.

Energy Northwest Response to 6.e:

The cost estimates for SAMA candidates presented in Table E.11-6 that were associated with plant equipment changes were minimal cost estimates that only used costs for design, licensing and NRC approval (where applicable), procurement, installation, testing, initial procedure development and initial training of maintenance or operations personnel. These estimates conservatively (including those identified in Table E11-2 from NUREG-1437 Supplements) did not include contingency costs for unforeseen implementation obstacles, replacement power for those that would require extended outages, or recurring training, maintenance and surveillance.

NRC Request:

f. The estimated implementation cost of $375,000 for SAMA AC/DC-23 seems high for what appears to be procedure development. Justify the cost estimate.

Energy Northwest Response to 6.f:

Typically, repair is not credited in the PSA unless it can be shown that the repair is simple and timely to accomplish (See Supporting Requirement DA-D9 in RG 1.200 Rev. 2). The breaker for which repair is to be modeled must be able to be completed in time to prevent core damage as a function of the accident sequence in which the breaker failure appears. For breakers that provide the pathway for restoration of the offsite or onsite power sources, the optimal repair time would be prior to depletion of the station batteries. For ECCS pump breakers, the optimal repair time could be within 30 minutes for high pressure injection or DHR. One type of repair scheme that accomplishes this is breaker replacement.

This SAMA adopts this principle and proposes that a 4160 Volt breaker failure could be repaired if roll-in spares were staged and ready for replacing the failed breaker. The cost for AC/DC-23 includes the cost for eight spare breakers, procedure development, engineering evaluation and staging restraints. The estimate assumed a feeder breaker that could be inserted for either an offsite or onsite supply and a load breaker for each major switchgear. In addition, a breaker was assumed to be shared by DG-1, DG-2, and DG-3. Since the switchgear is located in three different buildings, multiple staging locations are necessary, however, some switchgear would share the staged spare breakers due to reasonably close function and proximity (i.e., same building, redundant division).

For maximum flexibility the breakers would be procured Class 1E, Quality Class 1. Additionally, an engineering evaluation would be performed to assure the spare breaker could be used in the maximum number of cubicles of the switchgear to obtain maximum effectiveness and to support the development of procedures for the switchgear. The breakers would also be staged in locations where they could be seismically restrained but immediately available. They would be periodically rotated into service for surveillance testing.

- For SM-1 and SM-3, assume two shared breakers to replace the S1 or S3 breaker and a load breaker to the Division 1 or Division 2 emergency switchgear

- For SM-2, assume two breakers to replace the S2 breaker and the load breaker to SM-4

- For SM-7, and SM-8 assume two shared breakers, one to replace the supply breaker from SM-1(SM-3), the backup transformer, or the DG and the second to replace a load breaker to the largest ECCS or emergency SW pump

- For SM-4, assume one breaker for HPCS pump or the supply breaker from SM-2

- For the DGs, assume one breaker to be shared by DG-1, 2, and 3

The current manufacturer's cost for a typical Class 1 E, 4160 V, 1200 amp breaker is approximately $35,000 each for a total of $280,000. (Note: Some breakers would be as large as 3000 amps). Engineering evaluation and documentation cost are assumed to be $30,000 for breaker specification and design for seismic staging. Installation of staging restraints and setup of the breakers is assumed to be $45,000 total for 3 locations. New procedure development is $20,000. This provides a total cost estimate of $375,000.

The cost estimate did not include consideration of a contingency for the recurring costs for maintenance and surveillance testing of the extra breakers.

NRC Request:

g. The estimated cost for SAMA CC-03b is indicated to be $82K in ER Table E.11-6 and $160K in Table E-11-7. Clarify the discrepancy. Furthermore, either cost estimate seems high for what appears to be a minor software change. Justify the cost estimate.

Energy Northwest Response to 6.g:

The estimated cost for SAMA candidate CC-03b is $82,000. The cost estimate includes a required TS amendment to raise the RCIC backpressure allowable values to implement a higher setpoint and the existing switches would be used. Since the existing switches would be used, no cost associated with materials was incorporated into the estimate. The estimate included costs associated with engineering, licensing, NRC review, maintenance, training, and procedures.

The estimated cost for SAMA candidate CC-03b in Table E.11-7 was updated in Amendment 10 to correct the discrepancy.

NRC Request:

h. For certain Phase II SAMAs listed in Table E.11-6, the information provided does not sufficiently describe the associated modifications and what is included in the cost estimate. Provide a more detailed description of both the modification and the cost estimate for SAMAs AC/DC-27, CW-04, FR-07a, FR-07b, and HV-02.

Energy Northwest Response to 6.h:

a) AC/DC-27 examines the installation of permanent hardware changes that make it possible to establish 500 kV backfeed through the main step-up transformer. This modification provides a third potential source for offsite power supply to CGS, in addition to the Ashe and Benton lines. Backfeed was assumed to not be available for seismic analysis due to potential extensive losses of offsite power (implies a complete correlation between the 500 kV backfeed and the Ashe and Benton lines).

The cost estimate for AC/DC-27 includes design, licensing, load break switch procurement and installation of the load break switch between the 25 kV output of the turbine generator and the 25 kV isophase bus. Design modification of the isophase bus would also be required. CGS currently has manual disconnect links that require considerable time to remove before backfeed from the 500 kV grid can be brought back into the plant. Because of these time constraints, the 500 kV backfeed is not credited in the PSA. The cost estimate includes significant engineering and potential licensing actions associated with FSAR changes, including potential review by the NRC. Additionally, a reasonable estimate for installation was made. The cost break down was as follows:

Equipment (Load Break Switch only)	$500K
Engineering and Licensing	$800K
Installation	$400K
Total Cost	$1,700K

This cost estimate does not include additional training, procedures, and maintenance, throughout its life and is most likely considerably below the actual cost for this SAMA.

b) CW-04 – SAMA case replaces ECCS pump SW cooling with self-cooling seals. The only ECCS pump seals that would benefit from this SAMA are RHR A and B. They have SW cooled mechanical pump seals, and they are only required for shutdown cooling. The other LPCI pump (RHR C)

has already been evaluated and has had the SW cooling heat exchanger deactivated. LPCS, HPCS and RCIC do not have SW cooling for the pump seals.

The cost estimate for CW-04 is based on having self-cooling for the RHR A and B pump seals for use in normal shutdown cooling and when alternate shutdown cooling mode is used for core and suppression pool cooling during an accident. The cost estimate includes the following:

Engineering	$40K
Material	$380K
Installation	$200K
Ops training	$20K
Procedures	$15K
Licensing/NRC	$20K
Total Cost	$675K

c) FR-07a evaluates improvement of the fire resistance of cables to the containment vent valves. Fire damage to cables that support vent operation cause a failure of the containment venting function with probability 1.0 (containment vent has no redundant trains, and therefore fire damage that affects any portion of containment vent fails the vent function).

The cost estimate for FR-07a assumes that metal sheath cabling would be installed to replace the existing polymeric cabling. This type of installation has been performed at CGS and the estimate is based on past experience by the fire protection electrical engineer. Four circuits provide vent valve function. These circuits are safety-related Class 1E, Quality Class 1 circuits. The distance is significant and passes through numerous fire zones from the valves' power supplies to the control room and on to the reactor building vent valves. Each zone transition will require penetration and restoration of the zone's fire barriers. The cost will include design, engineering, installation and testing. Each circuit is estimated to cost $100K or $400K for all four circuits. The estimate does not include additional raceway or fire protection coatings. This cost estimate is based on a similar circuit installation at CGS.

d) FR-07b examines improvement of fire resistance of cables susceptible to disabling the TR-S transformer due to hot short. TR-S provides offsite power from the Ashe substation. The hot short probability was reduced to zero for the TR-S transformer.

The cost estimate for FR-07b assumes that metal sheath cabling would be installed to replace the existing cross linked polymeric cabling. There are two circuits that start and end at the same location that provide signals to the microwave control system at Ashe. These circuits provide signals that trip the 230 kV Ashe breaker to the plant's startup transformer. A hot short due to a fire could cause the trip signal to be falsely sent, tripping the primary credited offsite power source. These circuits have relatively short runs and pass through only a few fire zones. Additionally, they are non-safety related and their procurement can be commercial grade.

This type of installation of metal jacketed cabling has been performed at CGS for Appendix R cabling compliance in the past and the estimate is based on past experience by the fire protection electrical engineer for non-safety related circuits. The estimate is $50K per circuit, or $100K for both circuits.

e) HV-02 provides a redundant HVAC train or means of ventilation for the Division 1 and Division 2 switchgear rooms. This SAMA conceptualized that an existing Class 1E HVAC system for the cable spreading room and the cable chase could be used as a backup system to supply cooling to the Division 1 and 2 electrical switchgear and electrical equipment rooms (vital island). The conceptual design is to provide a Quality Class 1, Seismic Class 1 ventilation cross-connect ducting including appropriate isolation dampers to the vital island HVAC system. Since the HVAC for the cable spreading room and cable chase have redundant trains and only one train is required for cooling, the other train could be used as a backup to the vital island HVAC. The ducting would connect redundant division vital island HVAC trains and would be subject to NRC approval.

The cost estimate for HV-02 was estimated as follows:

Engineering	$100k
Material	$120K
Installation	$200K
Licensing	$40K
PSA	$20K
Total Cost	$480K

NRC Request:

i. ER Section E.12 discusses six sensitivity cases. Insufficient information is provided to understand the specific changes made to the baseline analysis assumptions for Cases 1 and 6. Provide a more detailed description of the analysis assumptions and methodology for these two cases.

Energy Northwest Response to 6.i:

Sensitivity Case 1 investigated the impact of assuming that damaged plant equipment is repaired and refurbished following an accident scenario. For this analysis, the cost of repair and refurbishment over the lifetime of the plant is equivalent to 20% of the replacement power cost in accordance with NUREG/BR-0184 "Regulatory Analysis Technical Evaluation Handbook" Section 5.7.6.3. The total benefit for Sensitivity Case 1 is calculated by taking the baseline total benefit subtracting the baseline replacement power costs and adding 20% of the baseline replacement power costs which represents the cost to repair and refurbish the plant.

Sensitivity Case 6 investigated the sensitivity of each analysis to the cost of replacement power. In order to determine the replacement power cost in 2008 dollars, the variable string power cost calculated in the baseline case was modified for the energy price inflation. The inflation rate was determined by assessing the electricity costs in 1993 and 2008. The retail electricity cost for the state of Washington in 1993 and 2008 were 3.65 and 6.69 cents/kW-h, respectively. The following equation was used to calculate the inflation rate.

$$z = \frac{2008\cos t}{1993\cos t} = \frac{6.69 cents/kW-h}{3.65 cents/kW-h} = 1.83$$

$$(1+x)^{(\Delta y)} = z$$

$$(1+x)^{(2008-1993)} = 1.83$$

$$x = 0.041 \Rightarrow 4.1\%$$

$$\mathrm{y} = \mathrm{year}$$

$$\mathrm{x} = \mathrm{inflation\ rate}$$

The final step calculated the 2008 value for the string of replacement power costs for CGS based on the calculated inflation rate. This value was used to calculate the replacement power costs in 2008 dollars.

$$B_{2008} = B_{1993}(1+0.041)^{(2008-1993)}$$
$$B_{2008} = (1.46E+08)(1+.041)^{(15)}$$
$$B_{2008} = \$\,2.68E+08$$

NRC Request:

j. ER Section E.12 states that "no explicit uncertainty was performed since the number of conservative assumptions and input account for any uncertainties in the calculations" and goes on to delineate several sources of conservatism in the SAMA analysis. This is not consistent with NEI 05-01 wherein the delineation of conservatisms is used to offset an uncertainty factor based on the 95th percentile CDF. Given that the 95th percentile values are typically a factor of two to five higher than point estimates, identify and provide a further evaluation of those SAMAs that are within a factor of two to five of being cost-beneficial. This evaluation can be based on more realistic estimates of risk reduction and implementation costs, and deterministic considerations, including potential negative implications of candidate SAMAs.

Energy Northwest Response to 6.j:

A sensitivity case was performed to assess the impact of using uncertainty factors for internal events, fire, and seismic based on the 95th percentile CDF for each analysis. The uncertainty factors were derived from the ratio of the 95th percentile to the mean point estimate for internal events, fire, and seismic CDF. Table 6.j-1 provides the uncertainty factors used. The results of the sensitivity case, shown in Table 6.j-2, found four SAMA candidates to be cost-beneficial for implementation at CGS. The cost-beneficial SAMA candidates are AC/DC-28, FR-07a, FR-07b, and CC-03b. Three of these SAMA candidates were also found to be cost-beneficial in the sensitivity case #2, which assessed the impact of a lower discount rate (see Table E.12-1). Therefore, using uncertainty factors based on the 95th percentile CDF confirmed the overall results of the SAMA analysis for the three previously identified SAMA cases and identified one additional SAMA candidate (CC-03b, raise RCIC back pressure trip set points).

These SAMA candidates will also be evaluated in the sensitivity study performed with the upgraded Revision 7.1 PSA model that is identified in the response to RAI 1.a. This sensitivity evaluation will include the use of the 95th percentile factor for CDF and provide the final assignment of the classification of cost-beneficial status.

Table 6.j-1: Uncertainty Factors	
	Uncertainty Factors
Internal Events	2.7
Fire	3.1
Seismic	3.2

Table 6.j-2: Uncertainty Factor Sensitivity Case Results				
SAMA ID	**Modification**	**Estimated Benefit**	**2008 Estimated Cost**	**Sensitivity Case Conclusion**
AC/DC-01	Provide additional DC battery capacity	$73,136	$1,799,200	Not Cost Effective
AC/DC-10	Provide an additional DG	$513,941	$10,816,000	Not Cost Effective
AC/DC-23	Develop procedures to repair or replace failed 4 kV breakers	$36,932	$375,000	Not Cost Effective
AC/DC-27	Install permanent hardware changes that make it possible to establish 500 kV backfeed through the main step-up transformer	$586,944	$1,700,000	Not Cost Effective
AC/DC-28	Reduce common cause failures between DG-3 and DG-1/2	$153,657	$100,000	Cost Effective
AC/DC-29	Replace DG-3 with a diesel diverse from DG-1 and DG-2	$320,883	$4,200,000	Not Cost Effective
AT-05	Add an independent boron injection system	$9,364	$800,000	Not Cost Effective
AT-07	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS	$0	$1,124,864	Not Cost Effective
AT-13	Automate SLC injection in response to ATWS event	$306	$660,000	Not Cost Effective
AT-14	Diversify SLC explosive valve operation	$613	$370,000	Not Cost Effective
CB-01	Install additional pressure or leak monitoring instruments for detection of ISLOCAs	$0	$5,600,000	Not Cost Effective

Table 6.j-2: Uncertainty Factor Sensitivity Case Results

SAMA ID	Modification	Estimated Benefit	2008 Estimated Cost	Sensitivity Case Conclusion
CB-03	Increase leak testing of valves in ISLOCA paths	$0	$400,000	Not Cost Effective
CB-08	Revise EOPs to improve ISLOCA identification	$0	$20,000	Not Cost Effective
CB-09	Improve operator training on ISLOCA coping	$0	$30,000	Not Cost Effective
CC-01	Install an independent active or passive high pressure injection system	$1,658,909	$29,120,000	Not Cost Effective
CC-02	Provide an additional high pressure injection pump with independent diesel	$1,658,909	$5,200,000	Not Cost Effective
CC-03b	Raise RCIC backpressure trip set points	$111,664	$82,000	Cost Effective
CC-20	Improve ECCS suction strainers	$0	$10,000,000	Not Cost Effective
CP-01	Install an independent method of suppression pool cooling	$929,862	$6,000,000	Not Cost Effective
CW-02	Add redundant DC control power for pumps	$44,748	$650,000	Not Cost Effective
CW-03	Replace ECCS pump motors with air-cooled motors	$184,528	$1,124,864	Not Cost Effective
CW-04	Provide self-cooled ECCS seals	$184,528	$675,000	Not Cost Effective
CW-07	Add an SW pump	$306,662	$6,136,000	Not Cost Effective
FR-03	Install additional transfer and isolation switches	$376,174	$2,000,000	Not Cost Effective
FR-07a	Improve the fire resistance of cable to the containment vent valve	$586,215	$400,000	Cost Effective
FR-07b	Improve the fire resistance of cable to transformer E-TR-S	$134,978	$100,000	Cost Effective
HV-02	Provide a redundant train or means of ventilation	$361,788	$480,000	Not Cost Effective
SR-03	Modify safety related CST	$0	$980,000	Not Cost Effective

NRC Request:

7) For certain SAMAs considered in the Environmental Report, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, provide an evaluation of the following SAMAs:

   a. Establishing procedures for opening doors and/or using portable fans for sequences involving room cooling failures, such as the emergency diesel generator room. Clarify if this is the intent for SAMA HV-03 or not.

Energy Northwest Response to 7.a:

HV-03 examined the procedures for loss of HVAC for any potential improvements. The review did consider the opening of doors and use of portable fans as potential improvements. The review found that the procedures did include steps to instruct operators to open doors and use portable fans when conditions favorable to these actions existed. The critical switchgear rooms, the ECCS pump rooms and the MCC rooms in the reactor building are specific areas where this alternate means of room cooling has been found effective and proceduralized. CGS performed thermal dynamic analysis of the compartments, where needed, to ensure adequate response time and that the alternate method of room cooling would be effective.

The DG room areas are of limited benefit from the approach of opening doors and installing fans. The engine/generator area is able to operate up to 130°F. However, the electrical equipment panel room immediately adjacent to the engine area is limited to 122°F. Ventilation flow paths are limited in the electrical equipment area to avoid drawing in higher engine room temperature.

These are the bases on which HV-03 in Table E.10-1 was screened as Criterion B "Already Implemented at CGS".

NRC Request:

   b. Utilizing a portable independently powered pump to inject into containment.

Energy Northwest Response to 7.b:

CGS has the capability and procedures to connect fire water for flooding via the condensate system into the containment via the vessel breach and to connect fire water for containment spray via a pumper truck through the fire water system. This is judged to meet the intent of the SAMA candidate proposed in the RAI.

NRC Request:

c. Using the security diesel generator and/or EDG-4 to extend the life of the 125-V DC batteries.

Energy Northwest Response to 7.c:

SAMA AC/DC-03 consists of constructing a permanent location for the portable DG-4 that can be aligned to either MC-7A or MC-8A, which would provide both AC power and DC power (i.e., through the battery charger) to the aligned train. The use of the security diesel would result in multiple use issues and require adding distribution equipment and cabling. DG-4 was specifically designed for this purpose. Enhancement of DG-4 to be more readily available is also the more cost effective SAMA option. Enhancement of DG-4 was evaluated to further increase its benefit in the response to RAI 6.b.

NRC Request:

d. Using a portable generator to provide power to individual 125-V DC MCCs upon loss of a DC bus to improve availability of HPCS.

Energy Northwest Response to 7.d:

The RAI suggests a means of improving DC power associated with HPCS by providing a portable DC generator. DC power is used in the HPCS system to provide power to flash the generator field, operate breakers remotely, and to operate the HPCS initiation logic including manual starting. All motor operated valves in the system are 480 Volt AC. When the HPCS system is operating on its DG's power, the DG-3 generator supplies the AC motive power including power to battery charger C1-7. The most likely cause of HPCS DC failure is the loss of AC power to the HPCS battery charger or charger failure. This SAMA would only be beneficial for scenarios where HPCS DG-3 output AC power is available with the loss of a HPCS DC charger or the battery. A portable DC generator to provide DC power to the HPCS DC circuit panel would require additional isolation hardware cabling most likely from DG-4 and potentially a back-up charger. The RRW of the HPCS DC system is less than 1.005 and it is judged that this SAMA would be of very little benefit and not cost beneficial.

Changed Section and Page Number		
Section Number	**Page Number**	**RAI number**
E.9.1	E-65	5b
E.9.2	E-68	5k
E.9.5	E-72	5f
Table E.3-2 Line Item Level 1	E-87	1e
Table E.5-1 Line Item 1	E-110	3g
Table E.9-3 Line Item AC/DC-29	E-143	5g
Table E.10-1 Line Item AC.DC-05	E-157	5m i
Table E.10-1 Line Item AC.DC-08	E-157	5m i
Table E.10 Insert A	E-157a	5m i
Table E.10 Insert B	E-157a	5m i
Table E.10 Line Item AC.DC-21	E-160	5m i
Table E.10 Insert A	E-160a	5m i
Table E.10 Line Item AC.DC-29	E-162	5m i
Table E.10 Line Item AT-04	E-162	5m i
Table E.10 Insert A	E-162a	5m i
Table E.10-1 Line Item CB-03	E-165	6b
Table E.10-1 Line Item CB-08	E-166	6b
Table E.10-1 Line Item CB-09	E-167	6b
Table E.10 Line Item CC-08	E-169	5m i
Table E.10 Line Item CC-09	E-169	5m i
Table E.10 Line Item CC-12	E-169	5m i
Table E.10 Insert A	E-169a	5m i
Table E.10 Insert B	E-169a	5m i
Table E.10 Insert C	E-169a	5m i
Table E.10 Line	E-170	5m i

Item CC-15		
Table E.10 Insert A	E-170a	5m i
Table E.10 Line Item CP-08	E-174	5m i
Table E.10 Insert A	E-174a	5m i
Table E.10 Line Item CP-17	E-176	5m i
Table E.10 Insert A	E-176a	5m i
Table E.10 Line Item FW-04	E-184	5m i
Table E.10 Insert A	E-184a	5m i
Table E.10 Line Item HV-05	E-185	5m i
Table E.10 Insert A	E-185a	5m i
Table E.10 line Item IA-05	E-187	5m i
Table E.10 Insert A	E-187a	5m i
Table E.10-1 Line Item SR-01	E-188	5k
Table E.11-7 Line Item CC-03b	E-215	6g

E.9 CANDIDATE SAMA IDENTIFICATION

The first step was to develop a comprehensive list of SAMA candidates to be subjected to the qualitative screening. The comprehensive list of SAMA candidates was developed by completing the following of tasks:

- Review of industry guidance documents and completed SAMA analyses.
- Review of the CGS IPE and IPEEE results.
- A review of the Level 1 PSA and Level 2 PSA results.
- Discussions with CGS personnel.

E.9.1 REVIEW OF INDUSTRY DATA

Since CGS is a BWR, particular interest was paid to existing SAMA candidates for BWRs. Nuclear Energy Institute (NEI) 05-01 [**2**] provides a standard list of BWR SAMA candidates, which was used as the starting point for the potential CGS SAMA candidates.

In addition to the SAMA candidates provided in Reference [**2**], Table 13, a review was undertaken of the BWR SAMA analyses completed and documented as supplements to NUREG-1437 [**54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65**]. These were reviewed to identify any SAMA candidates that might apply to CGS, but were not included in Reference [**2**]. ~~No additional candidates were identified by the review of the supplements to NUREG-1437.~~

Two additional candidates, AT-13 and AT-14, were identified based on the review of the supplements to NUREG-1437.

E.9.2 CGS IPE AND IPEEE REVIEW

A review was performed of the following documents:

- Individual Plant Examination Washington Nuclear Plant 2 Main Report, Revision 1, July 1994 [**66**].
- Individual Plant Examination of External Events Washington Nuclear Plant 2, Main Report Revision 0, June 1995 [**67**].

The IPE identified the major contributors to CDF for plant internal events, including internal floods. The IPE identified the following major contributors to plant CDF [**66**, Section 1.4.1]:

- LOOP (67%)

- SAMA analyses were performed to evaluate improving the resistance of cabling to the containment vent valve and to the 230kV start-up transformer (FR-07a and FR-07b).

- Seismic: The IPEEE [**67**, Section 1.4.1] states "The overall impression from the walkdowns and the review of the seismic qualification documentation is that the plant is well constructed and has a high resistance to seismic loading." The plant was conservatively designed for 0.25 g PGA, and most equipment was screened out. A specific evaluation was performed for MCCs. This evaluation determined the most limiting MCCs had a median capacity of 1.03g for anchoraged (0.44g HCLPF) and a median capacity of 1.00g for relay chatter (0.43 HCLPF). Conservative modeling (e.g., no recovery actions) resulted in a seismic CDF of 2.1E-05/year.

- The PSA seismic analysis calculates a seismic CDF of 5.25E-06/year, with the largest contributors being failures to primary containment and the reactor building. ~~Two~~ One seismic SAMA candidate~~s~~ ~~were~~ was evaluated. ~~Neither of these candidates was considered cost effective.~~

- Other External Events: Other external events (e.g., severe weather, external flooding, volcanic activity, and accidents at nearby facilities) were examined [**67**, Section 5]. Based on a progressive screening approach recommended in GL 88-20 [**70**], no significant vulnerabilities were identified and these events were screened from further evaluation [**67**, Sections 1.4.3 – 1.4.7]. Therefore, no SAMA candidates related to these external events were added to the list of potential SAMA candidates.

E.9.3 LEVEL 1 INTERNAL EVENTS DOMINANT CUTSETS

A review was performed of the top 100 cutsets for the Revision 6.2 of the CGS Level 1 PSA (internal events, including internal flooding) to identify the significant risk contributors. Table E.9-1 provides a summary of the top 100 Level 1 PSA core damage cutsets. This list of cutsets represents over 56% of the total CDF, and includes all cutsets individually contributing 0.1% or more of the total CDF.

From these cutsets the following significant contributors were identified:

- LOOP with CCF of all three EDGs.

  The initial SAMA candidate list included adding an additional diesel generator (AC/DC-10) and installing a gas turbine generator (AC/DC-15). Also, an additional SAMA (AC/DC-28) was evaluated to examine the benefits of reducing CCFs between the existing EDGs.

- SBO and failure of RCIC before power can be restored.

events and over one half of all core damage events resulting in release category LLN. Major contributors to release category LLN are:

- Initiating event followed by long-term loss of all ECCS injection : CC-01 and CC-02 (high pressure injection)
- Long-term loss of suppression pool cooling : CP-01 (additional suppression pool cooling training)
- Loss of high pressure injection and suppression pool cooling
- Long-term SBO : Numerous AC/DC SAMA candidates were examined including AC/DC-01, AC/DC-10, AC/DC-15, and AC/DC-28
- LOOP with long-term loss of high pressure injection and low pressure injection

SAMA candidates addressing all these contributors were evaluated.

LLS: Large, Late, Scrubbed Release (11.9%)

Release Category LLS is characterized by large failures of the containment that are located such that the release path passes through the suppression pool, thereby resulting in fission product scrubbing by the suppression pool water. Of the release categories modeled, this one is of lowest importance due to its smaller release and also its lower frequency of occurrence. Major contributors to Release Category LLS are:

- Reactor vessel rupture
- Initiating event followed by loss of high pressure injection and suppression pool cooling : CC-01 and CC-02 (high pressure injection)

SAMA candidates addressing high pressure injection and suppression pool cooling were already considered important because of the contribution of that combination of functional losses to LLN. No SAMA candidates to reduce the likelihood of vessel rupture were identified.

COK: Containment Intact (39.8%)

Approximately 40 percent of internal event core damage scenarios terminate with the containment intact. The core damage sequences that result in an intact containment with the highest frequency are:

- Long-term SBO with DC unavailable at the time of core melt and HPCS available.
- Short term SBO, with DC and ADS available at the time of core melt.
- Initiating event with short term loss of HPCS and ADS.
- Long-term SBO with DC not available at the time of core melt.

E.15 TABLES

Table E.3-1 Summary of CGS PSA

PSA	Documentation Revision Number	Date	Plant Mod	Data / Bayesian Update	Baseline CDF or LERF
Internal Events (including Internal Flooding) Level 1	3.0	8/2006	8/2006	6/2002	4.77E-6
Internal Events Level 2	2.0	1/2004	*	*	6.53E-7
Fire Level 1	2.0	11/2006	*	*	7.40E-6
Fire Level 2	2.0	11/2006	**	**	2.46E-7
Seismic Level 1	1.0	2/2007	*	*	5.25E-6
Seismic Level 2	1.0	2/2007	**	**	2.15E-6

* Plant Modifications and Data based on Internal Events Level 1 Model Revision 6.2
** Plant Modifications and Data based on Internal Events Level 2 Model Revision 6.2

Table E.3-2 Summary of CGS PSA Truncation Limits

	Fault Tree	Event Tree	Global
Internal Events	2E-9[3] to 1E-10		
Level 1	~~1E-10~~	7E-11[1] to 5E-12	5E-12
Level 2	1E-8 to 1E-14[2]	1E-13	1E-13
Fire			
Level 1	2E-9	1E-11	1E-11
Level 2	1E-8 to 1E-14[2]	1E-13	1E-13
Seismic			
Level 1	1E-10	1E-12	1E-12
Level 2	1E-12	1E-12	1E-12

[1] The quantification of six accident sequences is performed at approximately 7E-11 to maintain the number of cutsets for those sequences below a maximum set by the quantification program. All other event tree sequences are solved at a 5E-12 truncation.

[2] Depending on the fault tree, the truncation limit was adjusted to assure sufficient capture of the contributing basic events.

3. One fault tree was set at a truncation limit of 2E-9 to limit the large number of cutsets. This is approximately eight orders of magnitude lower than the fault tree solution. All other fault trees are solved at a 1E-10 truncation.

Table E.5-1 CGS Internal Events PSA Revision Records

Rev #	Issue Date	Revisions, Highlights, and Documentation	Results (/yr)
0	8/28/92	• Original submittal to NRC (GL 88-20 requirement) • Documented as WPPSS-FTS-133	CDF=5.42E-5 Level 2 (Release Frequency) = 5.09E-6
1	7/1994	• A request was made to NRC to discontinue reviewing the original submittal, and replaced it with this version as the GL 88-20 requirement. • Reassign this issuance to be Document WPPSS-FTS-133 • Major revisions performed in the following: 1. Common Cause Factor for SRVs, MSIVs, and circuit breakers 2. LOOP initiating frequency, event tree structure, and power recovery factors 3. HRA methodology 4. Enhanced MAAP calculations	CDF=1.75E-5 Level 2 (Release Frequency) = 1.07E-~~6~~ 5
2	8/1996	• In response to the NRC's RAI (First round has 39 questions, and second round has 3 questions), the following tasks were performed: 1. Updating the "Initiating Frequency", and developing a Failure Modes Effects Analysis (NE-02-94-36) 2. Adding the following Event Trees: • Loss of Div2 DC • Loss of AC Bus (SM1/2/3, SH5/6) • Loss of Control Room HVAC • Loss of SM-7 HVAC • Loss of SM-8 HVAC 3. Deleting the following Event Trees: • Loss of Service Water • Loss of CN (including Loss of CIA) 4. Adding RCIC as success path in the SORV event tree	CDF=1.43E-5 Level 2 (Release Frequency) did not update
	4/18/97	NRC issued IPE SER	
3	9/1997	• A major documentation enhancement and modeling improvement were performed for the BWROG PSA Certification Program. • The modeling improvements include the following: 1. Updating the "Test and Maintenance" unavailability rate using data up to 3/31/97. 2. Updating all random failure data using Bayesian method 3. Recalculating the CCF Data using Multiple Greek Letter Method 4. Revising the LOCA (large, medium, small) initiating frequency using EPRI/TR-102266 methodology with plant specific data 5. Recalculating the ISLOCA initiating frequency using NSAC-154 methodology Improving the TW sequences in all event trees	CDF=1.71E-5 Level 2 (Release Frequency) = 9.94E-6

Table E.9-3 List of Initial SAMA Candidates (continued)

	SAMA Description	Derived Benefit	System Importance (Num. Val. = RRW)	Source
AC/DC-29	Replace EDG-3 with a diesel diverse from EDG-1 and EDG-2.	A significant risk contributor to CGS is the CCF of EDG-1/2/3 to start. This SAMA would examine the benefit of replacing EDG-3 with a diesel of a different manufacturer from EDG-1 and EDG-2. CGS-specific SAMA candidate developed from PSA insights and input from CGS personnel.	1.44 (EDG Div. 1) 1.40 (EDG Div. 3 – HPCS) 1.27 (EDG Div.2)	Table E.9-1
Enhancements Related to ATWS Events				
AT-01	Create cross-connect ability for SLC trains.	Improved availability of boron injection during ATWS.	ATWS events comprise 30% of LERF.	[**2**, Table 13]
AT-02	Revise procedures to control vessel injection to prevent boron loss or dilution following SLC injection.	Improved availability of boron injection during ATWS.	ATWS events comprise 30% of LERF.	[**2**, Table 13], [**60**, Table G-4]
AT-03	Provide an alternate means of opening a pathway to the RPV for SLC injection.	Improved probability of reactor shutdown.	ATWS events comprise 30% of LERF.	[**2**, Table 13], [**57**, Table G-3]
AT-04	Increase boron concentration in the SLC system.	This will increase the time available for the operator to successfully initiated SLC.	ATWS events comprise 30% of LERF.	[**2**, Table 13], [**57**, Table G-3]
AT-05	Add an independent boron injection system.	Improved availability of boron injection during ATWS.	ATWS events comprise 30% of LERF.	[**2**, Table 13]
AT-06	Provide ability to use CRD or RWCU for alternate boron injection.	Improved availability of boron injection during ATWS.	ATWS events comprise 30% of LERF.	[**2**, Table 13], [**58**, Table G-4], [**59**, Table G-3]
AT-07	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Improved equipment availability after an ATWS.	ATWS events comprise 30% of LERF.	[**2**, Table 13]
AT-08	Increase SRV reseat reliability.	Reduced risk of dilution of boron due to SRV failure to reseat after SLC injection.	ATWS events comprise 30% of LERF.	[**2**, Table 13], [**58**, Table G-4], [**64**, Table G-5], [**65**, Table G-5]

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
AC/DC-04	Improve DC bus load shedding.	Criterion B Already Implemented at CGS	CGS operators are instructed by procedure to remove non-essential equipment to extend battery lifetime. Safety loads are separated from non-safety loads, making load shedding easy to accomplish. Therefore, the intent of this SAMA has already been implemented at CGS.	[80]
AC/DC-05	Provide DC bus cross-ties.	Criterion E Very Low Benefit	With the ability to provide alternate power from EDG-3 or EDG-4, this SAMA would provide little risk reduction. Therefore, this SAMA is not considered for further evaluation.	[81]
AC/DC-06	Provide additional DC power to the 120/240V vital AC system.	Criterion E Very Low Benefit	~~120/240 V AC is not risk significant at CGS. Therefore, this SAMA is determined to have a very low benefit and will not be considered.~~	[82]
AC/DC-07	Add an automatic feature to transfer the 120V vital AC bus from normal to standby power.	Criterion B Already Implemented at CGS	On loss of normal power, Divisions 1 and 2 Class 1E 120/240 V AC power is automatically transferred to standby AC power. Therefore, the intent of this SAMA has already been implemented at CGS.	[83]
AC/DC-08	Increase training on response to loss of two 120V AC buses that causes inadvertent actuation signals.	Criterion E Very Low Benefit	~~120/240 V AC is not risk significant at CGS. Therefore, this SAMA is determined to have a very low benefit and will not be considered.~~	[82]
AC/DC-09	Reduce DC dependence between high-pressure injection system and ADS.	Criterion B Already Implemented at CGS	ADS is dependent on Division 1 and Division 2 DC power while HPCS is dependent on Division 3 DC power. There is no DC dependence between Division 3 and Divisions 1 or 2 during operational conditions where ADS is required to be operable. Therefore, the intent of this SAMA has already been implemented at CGS.	[83]

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Insert B from Page E-157a

Insert A to Page E-157

120/240 V Critical Instrumentation Power System AC is not risk significant at CGS (no RRW values in Table E.9-2). Therefore, this SAMA candidate is determined to have a very low benefit and will not be considered.

Insert B to Page E-157

120/240 V Critical Instrumentation Power System AC is not risk significant at CGS (no RRW values in Table E.9-2). Therefore, this SAMA candidate is determined to have a very low benefit and will not be considered.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
AC/DC-20	Develop procedures for replenishing diesel fuel oil.	Criterion B Already Implemented at CGS	Each EDG has a minimum of seven days supply of fuel oil its associated storage tank. In addition, the auxiliary boiler storage tank is available as an additional source of diesel oil. Also, fuel oil can be delivered to the site within 12-24 hours from a remote source. Therefore, the intent of the SAMA has already been implemented at CGS.	[**38**, Section 9.5.4.1]
AC/DC-21	Use fire water system as a backup source for diesel cooling.	Criterion E Very Low Benefit	~~This would likely only be considered if power were lost to critical loads such as service water. In that case, only the diesel fire pump would be available, with limited inventory available. It is judged that for this scenario, the fire water inventory would be better used for core flooding, containment spray and other fuel cooling uses. Therefore, this SAMA is not considered for further evaluation.~~	Insert A from Page E-160a
AC/DC-22	Add a new backup source of diesel cooling.	Criterion E Very Low Benefit	This SAMA is similar in intent to AC/DC-21. Therefore, this SAMA is not considered for further evaluation.	
AC/DC-23	Develop procedures to repair or replace failed 4 kV breakers.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. Model Change – Make 4.16 kV breakers perfectly reliable in the PSA.	
AC/DC-24	In training, emphasize steps in recovery of off- site power after an SBO.	Criterion B Already Implemented at CGS	CGS procedures address restoration of off-site power in the event of degraded off-site power or an SBO event, with highest priority on restoration of off-site power. Therefore, the intent of the SAMA has already been implemented at CGS.	[**69**], [**81**]

Insert A to Page E-160a

This SAMA addresses a scenario where a loss of off-site power occurs along with a failure of service water. The service water system for all Level 1 events has RRW values of 1.06, 1.05, and 1.04. EDG are dominated by failure to start and failure to run from causes other than loss of service water. If a total loss of service water occurred, cooling would be lost to safety related components including RHR heat exchangers, LPCI and HPCS room coolers, and cooling to electrical switchgear rooms that provide power to these ECCS components would be lost. In that case, only the diesel fire pump would be available, with limited inventory available. It is judged that for this scenario, the fire water inventory would be better used for core flooding, containment spray, and other fuel cooling uses. Therefore, this SAMA is not considered for further evaluation.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

Insert - Table E.9-1

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
AC/DC-29	Replace EDG-3 with a diesel diverse from EDG-1 and EDG-2.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. Model Change – EDG-3 was removed from the common cause group of EDG-1 and EDG-2.	
Enhancements Related to ATWS Events				
AT-01	Create cross-connect ability for SLC trains.	Criterion B Already Implemented at CGS	SLC discharge piping is cross-tied to ensure full flow in the event that one squib valve fails. Therefore, the intent of this SAMA has already been implemented at CGS.	[**38**, Section 9.3.5.2]
AT-02	Revise procedures to control vessel injection to prevent boron loss or dilution following SLC injection.	Criterion B Already Implemented at CGS	The intent of this SAMA has already been implemented at CGS.	[**90**]
AT-03	Provide an alternate means of opening a pathway to the RPV for SLC injection.	Criterion B Already Implemented at CGS	CGS has the capability of injecting boron using the RCIC system. Therefore, the intent of the SAMA has already been implemented at CGS.	[**91**]
AT-04	Increase boron concentration in the SLC system.	Criterion E Very Low Benefit	~~Although this could provide some additional time for the operator to initiate SLC, the amount of time would not be extended significantly, due to the short time available to achieve shutdown. Therefore, this SAMA is not considered for further evaluation.~~	[**92**]

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Insert A to Page E-162a

The SLC system was not calculated to be risk significant per Table E.9-2. Although this could provide some additional time for the operator to initiate SLC, the amount of time would not be extended significantly, due to the short time available to achieve shutdown. Therefore, this SAMA is determined to have a very low benefit and will not be considered for further evaluation.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
AT-13	Automate SLC injection in response to ATWS event.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. This SAMA would increase the likelihood of initiating SLC injection by adding an automatic actuation in addition to the current manual actuation. Model Change – Make operator action to initiate SLC perfectly reliable.	
AT-14	Diversify SLC explosive valve operation.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. This SAMA would add diversity between the two SLC explosives valves to increase the reliability of SLC. Model Change – Remove CCF of the two SLC explosive valves.	
Enhancements Related to Containment Bypass				
CB-01	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. This SAMA would reduce the risk from ISLOCA events by providing early detection of leakage through interfacing systems.	
CB-02	Add redundant and diverse limit switches to each CIV.	Criterion E Very Low Benefit	Isolation at CGS is considered quite reliable. Therefore, this SAMA is not considered for further evaluation.	
CB-03	Increase leak testing of valves in ISLOCA paths.	~~Subsumed~~ ~~Subsumed by SAMA CB-01~~	Considered for a final cost-benefit evaluation. This SAMA would reduce the risk from ISLOCA events by providing early detection of leakage through interfacing systems.	

Criterion C
Considered for Further Evaluation

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
CB-04	Improve MSIV design.	Criterion B Already Implemented at CGS	Review of the IPE indicated that isolation failure of MSIVs was estimated to be 7.4E-4. CGS has initiated an extensive MSIV program, including installing improved solenoid valves and a modified preventative maintenance program with scheduled replacement for increased reliability.	[**66**]
CB-05	Install self-actuating CIVs.	Criterion E Very Low Benefit	Isolation at CGS is considered very reliable. Therefore, this SAMA is not considered for further evaluation.	
CB-06	Locate RHR inside containment.	Criterion D Excessive Implementation Cost	It is unlikely that RHR could be placed within primary containment. If possible, it is judged that the cost would be several million dollars. Therefore, this SAMA is not considered cost beneficial to implement at CGS.	
CB-07	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Criterion D Excessive Implementation Cost	The cost of implementing a similar SAMA at Vermont Yankee was estimated by Entergy Nuclear to require more than $2,500,000 in 2007. The cost associated with the implementation of this SAMA exceeds the attainable benefit for all SAMA candidates. Therefore, this SAMA is not considered cost beneficial to implement at CGS.	[**64**, Table G-5]
CB-08	Revise EOPs to improve ISLOCA identification.	~~Subsumed~~ ~~Subsumed by SAMA CB-01~~ Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. This SAMA would involve changes to the EOPs to improve ISLOCA identification. This SAMA would also involve additional operator training to cope with ISLOCAs.	

Criterion C
Considered for Further Evaluation

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
CB-09	Improve operator training on ISLOCA coping.	~~Subsumed~~ ~~Subsumed by SAMA CB-01~~	This SAMA is similar in intent to CB-08 and would involve implementing additional operator training in order to reduce the frequency of operator error while coping with ISLOCA events.	
Enhancements Related to Core Cooling System				
CC-01	Install an independent active or passive high pressure injection system.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. CGS has a high pressure injection pump with a dedicated diesel. The proposed modification would consist of adding redundant electric driven or steam driven high pressure injection pump to Division 2. Model Change – HPCS event tree functions were set to an unavailability of 1E-8.	
CC-02	Provide an additional high pressure injection pump with independent diesel.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. CGS has a high pressure injection pump with a dedicated diesel. The proposed modification would consist of adding redundant electric driven or steam driven high pressure injection pump to Division 2. Model Change – This model is quantified by Case CC-01.	
CC-03a	Raise HPCI backpressure trip set points.	Criterion A Not Applicable to CGS	CGS has a HPCS system instead of a HPCI system. The HPCS system uses a motor driven pump; therefore, backpressure trip does not apply. Therefore, the intent of the SAMA is not applicable to CGS.	[38]

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

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SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
CC-08	Modify ADS components to improve reliability.	Criterion E Very Low Benefit	~~The ADS system at CGS is very reliable and not risk significant. Therefore, this SAMA is not considered to be applicable to CGS.~~	[82]
CC-09	Add signals to open SRVs automatically in an MSIV closure transient.	Criterion E Very Low Benefit	~~For an MSIV closure event, one or more SRV's may open briefly. Opening of SRVs is very reliable and not a significant contributor to risk. Automatically opening the SRVs will increase the chance of failure to close and a resulting loss of RPV inventory. Therefore, this SAMA is not considered for further evaluation.~~	[82]
CC-10	Revise procedure to allow manual initiation of emergency depressurization.	Criterion B Already Implemented at CGS	Operators can manually initiate emergency depressurization when conditions so dictate, such as small LOCA with HPCS failure. Therefore, the intent of the SAMA has already been implemented at CGS.	[95]
CC-11	Revise procedure to allow operators to inhibit automatic vessel depressurization in non-ATWS scenarios.	Criterion B Already Implemented at CGS	Operators can inhibit automatic ADS in non-ATWS scenarios through procedural guidance. Therefore, the intent of the SAMA has already been implemented at CGS.	[90]
CC-12	Add a diverse low pressure injection system.	Criterion E Very Low Benefit	~~CGS has significant redundancy of low pressure systems, and they are not risk significant. Since, this SAMA is considered to be very low benefit for CGS it will not be considered. Therefore, this SAMA is not considered for further evaluation.~~	[82]

Insert B from page E-169a

Insert C from page E-169a

Insert A to Page E-169a

The ADS system at CGS has significant redundancy and high reliability. The resultant RRW is low and little improvement can be achieved. Therefore, this SAMA is determined to have a very low benefit and will not be considered for further evaluation.

Insert B to Page E-169a

SRVs at CGS have high redundancy (7 ADS + 11 SRVs) and high reliability. The resultant RRW is low and little improvement can be achieved. For an MSIV closure event, one or more SRV's may open briefly. Automatically opening the SRVs will increase the chance of failure to close and a resulting loss of RPV inventory. Therefore, this SAMA is determined to have a very low benefit and will not be considered for further evaluation.

Insert C to Page E-169a

CGS has significant redundancy of low pressure systems. The LPCI injection systems have RRW values of 1 and they are not risk significant. Therefore, this SAMA is determined to have a very low benefit and will not be considered for further evaluation.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
CC-13	Increase flow rate of suppression pool cooling.	Criterion E Very Low Benefit	The only impact identified for this SAMA would be to extend the time for the operators to initiate SLC during MSIV closure ATWS events. This operator action is already highly reliable and does not significantly contribute to risk. Therefore, this SAMA is not considered for further evaluation.	[**82**]
CC-14	Provide capability for alternate low pressure injection via diesel-driven fire pump.	Criterion B Already Implemented at CGS	CGS has the capability to use the fire protection water as a source for low pressure injection. This capability is credited in the CGS PSA. Therefore, the intent of the SAMA has already been implemented at CGS.	[**96**]
CC-15	Provide capability for alternate injection via RWCU.	Criterion E Very Low Benefit	~~RWCU has no source of water other than the RPV. It receives cooling from the TSW, therefore, if other sources of injection were unavailable, it is likely that RCWU would also be unavailable. Therefore, this SAMA is not considered for further evaluation.~~ Insert A from page E-170a	[**38**, Section 5.4.8.1]
CC-16	Revise procedure to align EDG to CRD pumps for vessel injection.	Criterion B Already Implemented at CGS	The CRD pumps are powered from Divisions 1 and 2, and are backed up by the respective EDGs. EOPs direct the use of the CRD pumps for injection when required. Therefore, the intent of the SAMA has already been implemented at CGS.	[**38**, Section 8.3] , [**97**]
CC-17	Revise procedure to allow use of condensate pumps for injection.	Criterion B Already Implemented at CGS	CGS has the capability to utilize condensate water as a source of low pressure injection. This capability is credited in the CGS PSA. Therefore, the intent of the SAMA has already been implemented at CGS.	[**97**]

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RWCU has no source of water other than the RPV. It receives cooling from the TSW, therefore, if other sources of injection were unavailable, it is likely that RCWU would also be unavailable. Therefore, this SAMA is determined to have a very low benefit and will not be considered for further evaluation.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
CP-08	Enhance procedures to refill CST from demineralized water or service water system.	Criterion E Very Low Benefit	Refilling the CST is already modeled in the PSA. CST inventory is not a significant contributor to RCIC unavailability. Therefore, this SAMA is not considered for further evaluation.	[92]
CP-09	Enhance procedure to maintain ECCS suction on CST as long as possible.	Criterion E Very Low Benefit	~~CST inventory is not a significant contributor to RCIC or HPCS unavailability. Therefore, this SAMA is not considered for further evaluation.~~	[103]
CP-10	Modify containment flooding procedure to restrict flooding to below the top of active fuel.	Criterion A Not Applicable to CGS	From PPM 5.7.1 "An accident in which the RPV is breached at an elevation below the top of the active fuel can be considered controlled only after the primary containment has been flooded to above the top of the active fuel." Therefore, the intent of the SAMA is not applicable to the CGS site.	[101]
CP-11	Install an unfiltered, hardened containment vent.	Criterion E Very Low Benefit	A sensitivity study performed as part of the CGS IPE concluded that a hardened vent would not significantly reduced off-site releases following core damage. Venting currently is an option for decay heat removal following loss of suppression pool cooling. Therefore, this SAMA is not considered for further evaluation.	[66]
CP-12	Install a filtered containment vent to remove decay heat. Option 1: Gravel Bed Filter Option 2: Multiple Venturi Scrubber	Criterion D Excessive Implementation Cost	The cost of implementing a similar SAMA at Vermont Yankee was estimated by Entergy Nuclear to require $3,000,000 in 2007. The cost associated with the implementation of this SAMA exceeds the attainable benefit for all SAMA candidates. Therefore, this SAMA is not considered cost beneficial to implement at CGS.	[64, Table G-5]

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RCIC and HPCS suctions are automatically switched from the CST to the suppression pool on a CST low level signal. Delaying the switchover would increase the likelihood of RCIS or HPCS failure due to insufficient NPSH, with little improvement to containment performance. Therefore, this SAMA is not considered for further evaluation.

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Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
CP-17	Improve wetwell–to–drywell vacuum breaker reliability by installing redundant valves in each line.	Criterion E Very Low Benefit	~~The wetwell to drywell vacuum breakers have been shown to not be risk significant at CGS. Therefore, this SAMA is not considered for further evaluation.~~	[**82**]
CP-18	Provide post-accident containment inerting capability.	Criterion E Very Low Benefit	The CGS containment is inerted at power conditions. The PSA quantifies hydrogen combustion as 5E-3. Therefore, this SAMA is not considered for further evaluation.	[**66**]
CP-19	Create a large concrete crucible with heat removal potential to contain molten core debris.	Criterion D Excessive Implementation Cost	The cost of implementing a similar SAMA at Vermont Yankee was estimated by Entergy Nuclear to require more than $100,000,000 in 2007. The cost associated with the implementation of this SAMA exceeds the attainable benefit for all SAMA candidates. Therefore, this SAMA is not considered cost beneficial to implement at CGS.	[**64**, Table G-5]
CP-20	Create a core melt source reduction system.	Criterion D Excessive Implementation Cost	The cost of implementing a similar SAMA at J.A. Fitzpatrick was estimated to cost more than $5,000,000. The cost associated with the implementation of this SAMA exceeds the attainable benefit for all SAMA candidates. Therefore, this SAMA is not considered cost beneficial to implement at CGS.	[**65**, Table G-5]
CP-21	Strengthen primary/secondary containment (e.g., add ribbing to containment shell).	Criterion D Excessive Implementation Cost	The cost of implementing a similar SAMA at Vermont Yankee was estimated by Entergy Nuclear to require more than $12,000,000 in 2007. The cost associated with the implementation of this SAMA exceeds the attainable benefit for all SAMA candidates. Therefore, this SAMA is not considered cost beneficial to implement at CGS.	[**64**, Table G-5]

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The wetwell-to-drywell vacuum breakers have been shown to not be risk significant at CGS based on the top cutsets and RRW values. Therefore, this SAMA is determined to have a very low benefit and will not be considered for further evaluation.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
FW-03	Install an independent diesel for the CST makeup pumps.	Criterion E Very Low Benefit	CGS has the ability to connect the diesel driven fire water pump to the suction of a condensate booster pump for RPV makeup. Therefore, it is judged that this SAMA would be of very low benefit. Therefore, this SAMA is not considered for further evaluation.	[96]
FW-04	Add a motor-driven feedwater pump.	Criterion E Very Low Benefit	~~A motor-driven reactor feedwater pump would still be dependent on the lower pressure Condensate and Condensate booster pumps for NPSH. Therefore, this SAMA is not considered for further evaluation.~~	[82]
Enhancements Related to Heating, Ventilation, and Air Conditioning				
HV-01	Provide reliable power to control building fans.	Criterion B Already Implemented at CGS	EDG backed power is provided to control building fans that serve the control room, cable spreading room, critical switchgear rooms and remote shutdown room. Therefore, the intent of this SAMA has already been implemented at CGS.	[38, Table 8.3-1]
HV-02	Provide a redundant train or means of ventilation.	Criterion C Considered for Further Evaluation	Considered for a final cost-benefit evaluation. Simultaneous loss of cooling to both critical switchgear rooms is a significant contributor to risk. Model Change – Remove the switchgear dependency on HVAC and eliminate the loss of HVAC initiating event. Loss of switchgear HVAC IE sequences were set to zero.	

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Based on the system fault tree modeling, reactor feedwater pumps were not a dominant contributor (<20%) of failure of the reactor feedwater pumps. A motor-driven reactor feedwater pump would still be dependent on the low pressure condensate and condensate booster pumps for NPSH. Therefore, this SAMA is not considered for further evaluation.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
HV-03	Enhance procedures for actions on loss of HVAC.	Criterion B Already Implemented at CGS	CGS procedures address loss of ventilation in the Turbine Building, Reactor Building, Control Room, cable Spreading Room, Critical Switch Gear Room and Remote Shutdown Rooms. . Therefore, the intent of the SAMA has already been implemented at CGS.	[**109**]
HV-04	Add a diesel building high temperature alarm or redundant louver and thermostat.	Criterion B Already Implemented at CGS	Temperature sensors in the diesel generator rooms and in the exhaust ducts annunciate alarms in the event of abnormally high or low temperatures. Therefore, the intent of the SAMA has already been implemented at CGS.	[**38**, Section 9.4.7.5]
HV-05	Create ability to switch HPCS and RCIC room fan power supply to DC in an SBO event.	Criterion E Very Low Benefit	~~Room cooling is not required for RCIC. If electric power is unavailable to HPCS room cooling, it is highly likely that electric power would be unavailable to HPCS components. Therefore, this SAMA is not considered for further evaluation.~~ Insert A from page E-185a	[**38**]
HV-06	Enhance procedure to trip unneeded RHR or core spray pumps on loss of room ventilation.	Criterion E Very Low Benefit	Each ECCS pump is located in a separate room. Each room has a room cooler with fans powered from the associated division and cooling water supplied by the respective divisions of SSW. Failures in the HVAC of one division would not impact the operability of components in the other divisions. Therefore, this SAMA is not considered for further evaluation.	[**38**, Section 9.2]
HV-07	Stage backup fans in switchgear rooms.	Criterion B Already Implemented at CGS	The switchgear rooms have staged backup fans. Therefore, the intent of the SAMA has already been implemented at CGS.	[**109**]

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Room cooling is not required for RCIC. If electrical power is unavailable to HPCS room cooling, it is highly likely that electric power would be unavailable to HPCS components. In addition, procedures direct operators to implement contingency measures such as opening doors and using portable fans in the event of loss of HVAC. Therefore, this SAMA is not considered for further evaluation.

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Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
IA-05	Improve SRV and MSIV pneumatic components.	Criterion E Very Low Benefit	~~SRVs and MSIVs are very reliable, and further improvement would not contribute significantly to plant risk. Therefore, this SAMA is not considered for further evaluation.~~	[82]
Other Enhancements				
OT-01	Install digital large break LOCA protection system.	Criterion E Very Low Benefit	Large LOCA is not a large risk contributor, and this modification is not considered to significantly reduce the risk of a large LOCA. Therefore, this SAMA is not considered for further evaluation.	[82]
OT-02	Enhance procedures to mitigate large break LOCA.	Criterion E Very Low Benefit	Large break LOCAs are dominated by automatic initiation of mitigating systems. Operator actions are not significant contributors. Therefore, this SAMA is not considered for further evaluation.	[66]
OT-03	Install computer aided instrumentation system to assist the operator in assessing post-accident plant status.	Criterion B Already Implemented at CGS	The CGS Safety Parameter and Display System (SPDS) and Graphics Display System (GDS) provide status of plant safety functions and support information for emergency response. Therefore, the intent of this SAMA has already been implemented at CGS.	[110]
OT-04	Improve maintenance procedures.	Criterion E Very Low Benefit	No plant maintenance procedures have been identified as being significant contributors to plant risk. Therefore, this SAMA is not considered for further evaluation.	[66]
OT-05	Increase training and operating experience feedback to improve operator response.	Criterion E Very Low Benefit	No plant training or feedback issues have been identified as being significant contributors to plant risk. Therefore, this SAMA is not considered for further evaluation.	[66]

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SRVs and MSIVs are very reliable based on top cutsets and RRW values. Further improvement would not contribute significantly to plant risk. Therefore, this SAMA is not considered for further evaluation.

Table E.10-1 Qualitative Screening of SAMA Candidates (continued)

SAMA ID	Modification (Potential Enhancement)	Screening Criterion	Basis for Screening/ Modification Enhancements	Source
OT-06	Develop procedures for transportation and nearby facility accidents.	Criterion B Already Implemented at CGS	CGS has procedures to address accidents on the Hanford site and shipping accidents. Therefore, the intent of this SAMA has already been implemented at CGS.	[110]
Enhancements to Reduce Seismic Risk				
SR-01	Increase seismic ruggedness of SSW pumps and RHR heat exchangers.	E Criterion ~~G~~ ~~Considered for Further Evaluation~~ Very Low Benefit	This SAMA candidate was screened as "Considered for Further Evaluation" (Criterion C) on the basis that sequence SDS41S01 contributes 15.6% to the total seismic CDF. CDF results from seismic failure of SSW pumps and RHR heat exchangers. Typically a "Criterion C" screened SAMA candidate would be evaluated by obtaining the delta CDF, delta-release category vector, and cost of implementation. However, this SAMA was treated differently. In subsequent discussions, it was apparent that a qualitative argument existed to reconcile this SAMA candidate. The sequence of SDS 41 includes loss of piping, DC panels, and MCR panels (relays), in addition to losses of RHR heat exchangers or the SSW pumps. It was concluded that just strengthening RHR heat exchangers and service water pumps would not be beneficial. On this qualitative basis, it was not necessary to pursue the CDF/cost of implementation approach, and this SAMA was not considered for further evaluation on the basis of low benefit.	

Table E.11-7 Final Results of the Cost Benefit Evaluation (continued)

SAMA ID	Modification	Analysis Cases	Estimated Benefit	2008 Estimated Cost	Conclusion
CB-09	Improve operator training on ISLOCA coping.	Case 09	$0	$30,000	Not Cost Effective
CC-01	Install an independent active or passive high pressure injection system.	Case 10	$875,249	$29,120,000	Not Cost Effective
CC-02	Provide an additional high pressure injection pump with independent diesel.	Case 11	$875,249	$5,200,000	Not Cost Effective
CC-03b	Raise RCIC backpressure trip set points.	Case 12	$53,740	~~$160,000~~ $82,000	Not Cost Effective
CC-20	Improve ECCS suction strainers.	Case 13	$0	$10,000,000	Not Cost Effective
CP-01	Install an independent method of suppression pool cooling.	Case 15	$541,841	$6,000,000	Not Cost Effective
CW-02	Add redundant DC control power for pumps.	Case 18	$24,618	$650,000	Not Cost Effective
CW-03	Replace ECCS pump motors with air-cooled motors.	Case 19	$105,916	$1,124,864	Not Cost Effective
CW-04	Provide self-cooled ECCS seals.	Case 19	$105,916	$675,000	Not Cost Effective
CW-07	Add a service water pump.	Case 20	$177,704	$6,136,000	Not Cost Effective
FR-03	Install additional transfer and isolation switches.	Case 21	$208,943	$2,000,000	Not Cost Effective
FR-07a	Improve the fire resistance of critical cables.	Case 22	$333,703	$400,000	Not Cost Effective
FR-07b	Improve the fire resistance of critical cables.	Case 22a	$75,446	$100,000	Not Cost Effective
HV-02	Provide a redundant train or means of ventilation.	Case 23	$211,659	$480,000	Not Cost Effective
SR-03	Modify safety related CST.	Case 25	$0	$980,000	Not Cost Effective